

# **Appendix 1.0**

## **Next Generation Nuclear Plant**

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## A1.1 INTRODUCTION AND BACKGROUND

In the approaching decades, the United States, the other industrialized countries, and the entire world will need energy and an upgraded energy infrastructure to meet the growing demands for electric power and transportation fuels. Anticipating this critical need, the Generation IV International Forum (GIF) identified nuclear energy system concepts for producing electricity that excel at meeting the goals of superior economics, safety, sustainability, proliferation resistance, and physical security. One of these concepts—the Very-High-Temperature Gas-Cooled Reactor (VHTR)—is uniquely suited for producing hydrogen without consuming fossil fuels or emitting greenhouse gases. Working with its international partners in the Generation IV International Forum, the U.S. Department of Energy is conducting research and development (R&D) on this system for the Next Generation Nuclear Plant (NGNP) project, a project to demonstrate emissions-free nuclear-assisted electricity and hydrogen production by 2017.

This Plan addresses three aspects of NGNP research and development that are imperative to achieving demonstration of the NGNP by 2017: (1) fuel development and qualification, (2) materials testing and qualification, and (3) design methods development and evaluation. The Plan does not address hydrogen production or energy conversion technologies, which are addressed in separate parallel programs.

This plan provides an overview of the planning for research and development for the NGNP. For additional detail, see the *Next Generation Nuclear Plant Research and Development Program Plan*, INEEL/EXT-05-02581, December 2004.

### A1.1.1 System Description

The NGNP Generation IV reference concept is a helium-cooled, graphite moderated, thermal neutron spectrum reactor with an outlet temperature of 900 to 1000°C. Although final selection of the reactor core technology will not be made until after the pre-conceptual design phase of the project, the reactor core is envisioned to either be a prismatic graphite block or pebble bed gas-cooled reactor. A liquid-salt cooled reactor may also be considered. The NGNP will produce both electricity and hydrogen. The process heat for the hydrogen production, and possibly the electricity production, will be transferred through an intermediate heat exchanger (IHX). The reactor thermal power and core configuration will be designed to assure passive decay heat removal without fuel damage during any hypothetical accident. The fuel cycle will be a once-through, very high burnup, low-enriched uranium fuel cycle. Figure A1.1 provides an example schematic of one possible design for the NGNP.

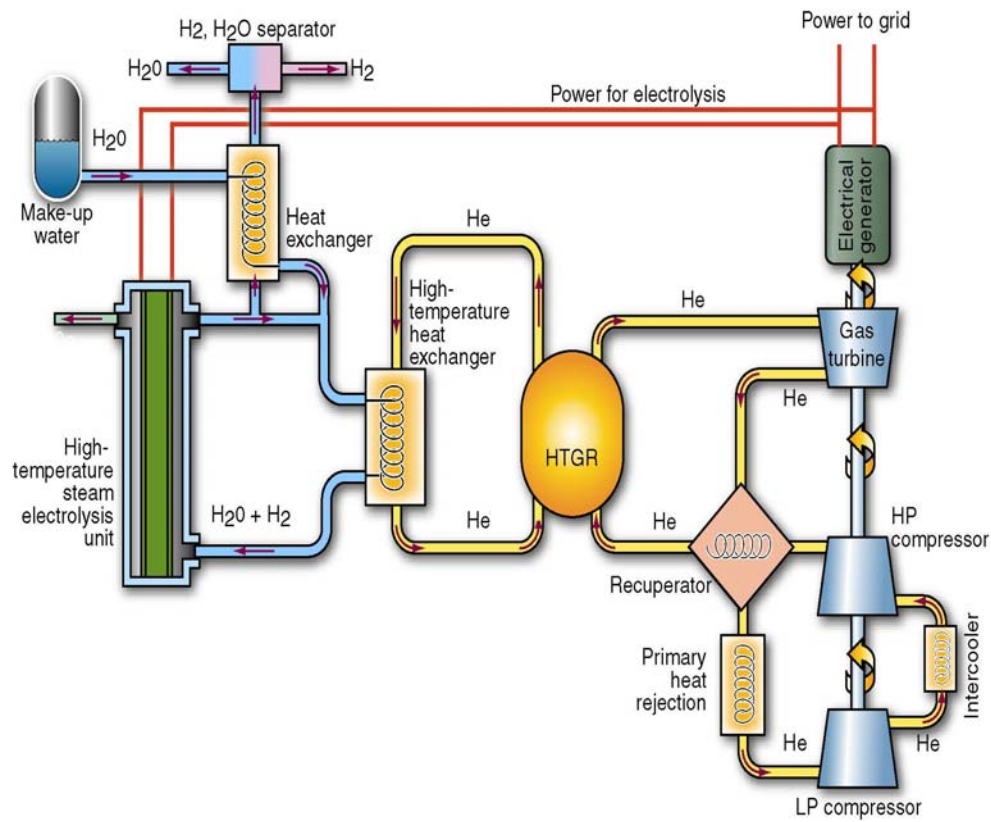


Figure A1.1. Schematic of one possible NGNP configuration.

One or more processes will use the heat from the high temperature helium coolant to produce hydrogen. The first process of interest is the thermo-chemical splitting of water into hydrogen and oxygen. The second process of interest is thermally assisted electrolysis of water.

The Gas Turbine-Modular Helium Reactor (GT-MHR) system shown in Figure A1.2 is a prismatic block gas-cooled reactor. The Prismatic block version of a VHTR may be similar to the GT-MHR except the VHTR would be designed to operate at a higher temperature.

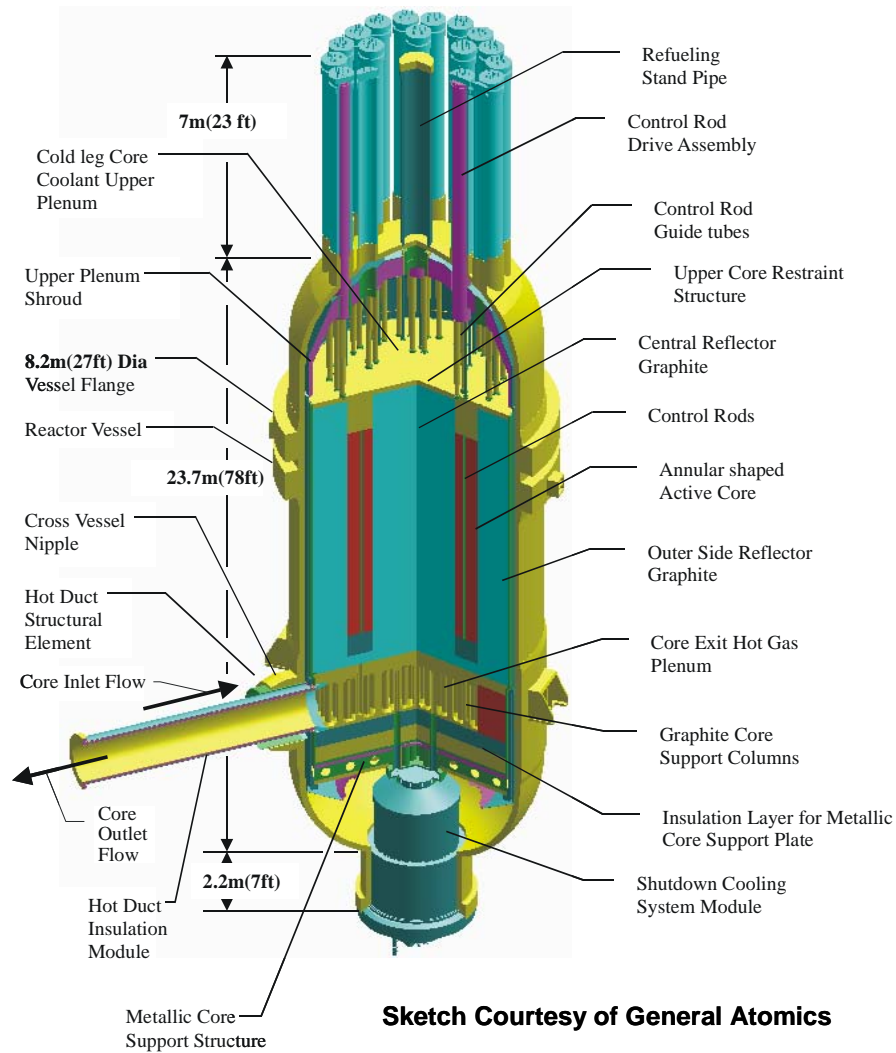


Figure A1.2. GT-MHR reactor system cutaway diagram.

## A1.1.2 Overall System Timeline

A target schedule for the development and construction of the NGNP includes starting with pre-conceptual designs in 2006, Preliminary Design completion in 2009, completion of major R&D activities by 2012, and Construction starting in 2012. NGNP Operations are scheduled to begin in 2017.

## A1.2 RESEARCH AND DEVELOPMENT STRATEGY

### A1.2.1 Objectives

The major objectives for the NGNP are:

- Demonstrate a full-scale prototype VHTR, or other commercially viable reactor technology, that is licensed by the U.S. Nuclear Regulatory Commission
- Demonstrate safe and economical nuclear-assisted production of hydrogen and electricity.

It is expected that a successful NGNP demonstration will lead to widespread deployment of VHTRs in the U.S. and elsewhere. The performance goals for an “nth of a kind” VHTR plant are as follows:

- Plant overnight construction cost: <\$1,000/kW
- Electricity generation cost: <1.0 cents/kW-hr
- Hydrogen cost: <\$1.50/gallon – gasoline equivalent
- Development of a system that meets or exceeds NRC regulatory requirements when built on a commercial site.

### **A1.2.2      Scope**

The Department of Energy (DOE) published a draft strategy for developing and demonstrating the NGNP on May 27, 2004. Informed by public comment on the draft strategy, the DOE is now formulating its final acquisition strategy. The strategy will be designed to maximize industrial and international cost-shared participation.

The preliminary rough order of magnitude cost estimate range for the NGNP is \$1.8 to 2.4 billion. The scope of the work will be divided into five phases with provisions for go/no-go decisions at the end of each phase:

- Phase 1: Project Integration and Formation
- Phase 2: Research and Preliminary Design
- Phase 3: Development and Final Design
- Phase 4: Construction and Viability Testing
- Phase 5: Project Close-out

The following tasks will be completed in Phase 1:

- Conduct a design competition for the NGNP reactor technology and select the NGNP technology
- Prepare a “Research and Development Plan” that supports the selected technology
- Prepare a “Fuel Plan” detailing the acquisition of licensed fuel for the NGNP
- Prepare a “Business Plan” that details the successive phases of the project and identifies the members of the international consortium that will cost share the project and lead its development.

The DOE will be substantially involved in the technology selection process and have ultimate approval. Quarterly reporting will be required in accordance with 10 CFR 600. The program will be run consistent with DOE Order 413.3.

The DOE laboratory system, led by the Idaho National Laboratory (INL), will be the principal resource for conducting the NGNP R&D. The laboratories will perform R&D that will be critical to the success of the NGNP, primarily in the areas of:



- High temperature gas reactor fuels behavior
- High temperature materials qualification
- Design methods development and validation
- Hydrogen production technologies
- Energy conversion

The current R&D work is addressing fundamental issues that are relevant to a variety of possible NGNP designs. When the NGNP technology is selected, the R&D will be aligned with the selected technology. This appendix describes the NGNP R&D planned and currently underway in the first three topic areas listed above. The energy conversion technologies program is described in Appendix 8. The DOE-funded hydrogen production is described elsewhere.

#### **A1.2.2.1 High Temperature Gas Reactor Fuels Behavior**

The DOE-NE Advanced Gas Reactor (AGR) Fuel Development and Qualification Program is designed to provide a fuel qualification baseline in support of NGNP licensing and operation. The AGR Fuel Development and Qualification Program will also support the near-term deployment of VHTRs for commercial energy production in the United States by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification. The activities conducted under this area are also largely applicable to non-gas, very high-temperature fuel designs.

The program will: (1) develop technologies for the manufacture of very high quality fuel kernels, TRISO-coated particles, and compacts; (2) irradiate fuel to high burnup at prototypical powers; (3) test the irradiated fuel during worst-case accident simulations, and (4) develop and validate physically based computer models of the fuel and fission product transport behavior.

#### **A1.2.2.2 High Temperature Materials Qualification**

The NGNP Materials R&D Program will focus on testing and qualification of the key materials commonly used in very high-temperature designs. The materials R&D Program will address the materials needed for the NGNP reactor, power conversion unit, intermediate heat exchanger, and associated balance of plant. The DOE Office of Nuclear Energy's Nuclear Hydrogen Initiative (NHI) will address materials for hydrogen production. The order of priority for the NGNP materials R&D is as follows:

- Test and qualify core graphite materials
- Develop an improved high temperature design methodology for use of selected metals at very high temperatures
- Develop American Society of Mechanical Engineers (ASME) and American Society for Testing and Materials (ASTM) codes and standards
- Perform environmental testing and thermal aging of selected high temperature metals
- Reactor Pressure Vessel (RPV) materials irradiation testing and qualification
- Development and qualification of composites for use in control rod cladding and guide tubes
- Resolve RPV fabrication and transportation issues

### **A1.2.2.3 Design Methods Development and Validation**

The Methods Development and Validation effort focuses on the development of tools to assess the neutronic and thermal-hydraulic behavior of the plant. Fuel behavior and fission product transport models are also being developed in the AGR Fuel Program. In addition, various stress analyses and mechanical design tools will also need to be developed and validated. The Methods Development and Validation process includes scenario identification, defining the phenomena identification and ranking tables (PIRT), completing the required development (for models and other tools), and performing the necessary validation studies.

### **A1.2.2.4 Quality Assurance**

All work performed to support the NGNP R&D Program will be in accordance with *The Next Generation Nuclear Plan (NGNP) Quality Assurance Program*, INEEL/EXT-04-01776, and will utilize the national consensus standard ASME NQA 1997, "QA Program Requirements for Nuclear Facilities Applications," and Subpart 4.2 of ASME NQA 2000, "Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development."

## **A1.2.3 Viability Issues**

No viability issues are associated with the VHTR. The basic technology for the NGNP has been established in the former High Temperature Gas Reactor test and demonstration plants (DRAGON, Peach Bottom, AVR, Fort St. Vrain, and THTR). In addition, the technologies for the NGNP are being advanced in the Gas Turbine-Modular Helium Reactor (GT-MHR) Project, and the South African state utility ESKOM sponsored project to develop the Pebble Bed Modular Reactor (PBMR). Furthermore, the Japanese HTTR and Chinese HTR-10 test reactors are demonstrating the feasibility of some of the planned NGNP components and materials.

## **A1.2.4 Research Interfaces**

A key to the cost effective long-term development, testing, and demonstration of the NGNP will be strategic partnerships that in combination will provide the needed talents, infrastructures, and resources for much of the R&D. Of special importance are partnerships of the DOE and INL with industry, researchers at universities and national laboratories, and other GIF countries.

### **A1.2.4.1 Relationship to GIF R&D Projects**

The GIF has established a "VHTR System Steering Committee" for coordination of the VHTR R&D. The GIF VHTR member countries are USA, France, Japan, United Kingdom, Korea, Canada, Switzerland, South Africa, and Euratom. Project Management Boards, reporting to the Steering Committee, have been established to define R&D collaborations in specific areas. Four Project Management Boards are now active for the VHTR-- Materials and Components, Fuel and Fuel Cycle, Hydrogen Production, and Design and Safety Methods.

Each Project Management Board will develop multiple collaboration agreements within their area. For example, the Materials and Components Board is developing collaboration agreements for:

1. Graphite development and qualification
2. Composites (carbon-carbon, SiC-SiC), and

### 3. Vessel steel qualification.

The member countries of the forum are expected to sign a framework agreement (government to government) during FY 2005, which will provide the legal agreements allowing the productive, yet protected, sharing of research and development results.

#### **A1.2.4.2 University Collaborations**

Partnerships with universities will be conducted through the Nuclear Energy Research Initiative (NERI) process and formal sub-contracts. The nature of these partnerships will focus on peer-reviewed, investigator-led projects in academia, as well as program R&D tasks jointly undertaken by universities and the DOE laboratories.

#### **A1.2.4.3 Industry Interactions**

The NGNP strategy is designed to engage private sector and international (e.g., agents of countries that are members of the GIF) cost-shared participation in this effort, while also integrating the research work to be conducted at the INL and other DOE laboratories. One management and funding option that the DOE is considering is the feasibility and effectiveness of entering into a cost-shared, cooperative agreement with an industrial partner to lead the formation of an international consortium and integrate and execute the activities associated with this project.

#### **A1.2.4.4 I-NERI**

A number of International-NERIs (I-NERI) are planned throughout the life of the NGNP R&D Program, some of which are currently in negotiations and others that are only envisioned. Table A1.1 provides a summary of I-NERIs currently starting or in progress.

Table A1.1. NGNP related I-NERIs.

<b>I-NERI Title</b>	<b>Collaborators</b>	<b>Scope</b>	<b>Period</b>
<b>Fuels</b>			
Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels	INL, French Centre d'Etude Atomique (CEA), Massachusetts Institute of Technology (MIT)	Develop improved fuel behavior models for gas reactor coated particle fuels. Develop improved coated-particle fuel designs that can be used reliably at very high burnups.	2003-2006
<b>Materials</b>			
Composites	INL, Oak Ridge National Laboratory (ORNL), French CEA	Develop, irradiate, and qualification test tubular SiC/SiC composite material for control rods	2005-2010
<b>Design Methods and Evaluation</b>			
Development of Safety Analysis Codes and Experimental Validation for a Very-High-Temperature Gas-Cooled Reactor	INL, University of Michigan, Korea Advanced Institute of Science & Technology (KAIST) and Seoul National University	Develop new Advanced Computational Methods for safety analysis codes and numerical & experimental validation of these computer codes. Includes improving two well-respected light water reactor transient response codes (RELAP5/ATHENA and MELCOR).	2005-2006
Thermal-Hydraulic Analyses And	INL, Argonne National Laboratory (ANL), Iowa	Provide benchmark data for the assessment and improvement of thermal-hydraulic	2005-2007

<b>I-NERI Title</b>	<b>Collaborators</b>	<b>Scope</b>	<b>Period</b>
Experiments for VHTR Safety	State University, Stanford University and French Investigating Organization: CEA	codes proposed for evaluating normal and reduced power operations plus decay heat removal concepts and designs in VHTRs.	
Screening of Gas-Cooled Reactor Thermal-Hydraulic and Safety Analysis Tools and Experiment Database	ANL, INL, and Korean Investigating Organization: Korea Atomic Energy Research Institute	Develop a formal qualification framework, (b) an initial filtering of the existing databases, and (c) a preliminary screening of tools for use in thermal-hydraulics and safety analyses.	2005-2006

## **A1.3 HIGHLIGHTS OF R&D**

### **A1.3.1 Fuel Development**

VHTRs in the United States will focus on cost competitive designs employing inherently safe ceramic cores, with the TRISO coated particle providing the primary containment to prevent release of fission products. The DOE AGR Fuel Development and Qualification Program is designed to provide a fuel qualification baseline with the following overall goals:

- Provide a baseline fuel qualification data set in support of the licensing and operation of the NGNP. Gas-reactor fuel performance demonstration and qualification comprise the longest duration R&D task for NGNP feasibility. The baseline fuel form is to be demonstrated and qualified for a peak fuel centerline temperature of 1250°C.
- Support near-term deployment of VHTRs by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Utilize international collaboration mechanisms to extend the value of DOE resources.

The AGR Fuel Development and Qualification Program is operated under the joint management (co-lead management and coordination) of the INL and ORNL. The Program consists of five elements:

1. Fuel manufacture,
2. Fuel and materials irradiations and post irradiation examinations (PIE),
3. Safety testing,
4. Fuel performance modeling, and
5. Fission product transport and source term modeling.

An underlying theme for the fuel development work is the need to develop a more complete fundamental understanding of the relationship between the fuel fabrication process, key fuel properties, the irradiation performance of the fuel, and the release and transport of fission products in a VHTR primary coolant system. Fuel performance modeling and analysis of the fission product behavior in the primary circuit are considered essential to the plant designer in establishing the core design and operating limits and demonstration to the licensing authority that the applicant has a thorough understanding of the

in-service behavior of the fuel system. The fission product behavior task will also provide primary source term data needed for licensing.

#### A1.3.1.1 Fuel Form

The fuel for the NGNP is based on the TRISO-coated particle fuel design (see Figure A1.3) demonstrated in high-temperature gas-cooled reactors in the United Kingdom, United States, Germany, and elsewhere. The TRISO-coated particle is a spherical-layered composite about 1 mm in diameter. It consists of a kernel of uranium dioxide ( $\text{UO}_2$ ) or uranium oxycarbide (UCO) surrounded by a porous graphite buffer layer that absorbs radiation damage, allows space for fission gases produced during irradiation, and resists kernel migration at high temperatures. Surrounding the buffer layer are a layer of dense pyrolytic carbon called the IPyC, a silicon carbide (SiC) layer, and a dense outer pyrolytic carbon layer (the OPyC). The pyrolytic carbon layers shrink under irradiation and create compressive forces that act to protect the SiC layer, which is the primary pressure boundary for the microsphere. The inner pyrolytic carbon layer also protects the kernel from corrosive gases present during deposition of the SiC layer. The SiC layer is also the primary containment of fission products generated during irradiation and under accident conditions. Each microsphere acts as a mini pressure vessel. The baseline fuel kernel for the NGNP is low-enriched (about 15% U-235) UCO instead of  $\text{UO}_2$ , owing to performance issues associated with the  $\text{UO}_2$  fuel at high power and burnups. At the high power densities expected in the NGNP ( $>6 \text{ W/cm}^3$ ), the associated large thermal gradients can drive kernel migration in  $\text{UO}_2$ -coated particles. The fission products in a  $\text{UO}_2$  fuel kernel that moves through the buffer and inner pyrocarbon layers may damage the SiC layer. Furthermore, at the high burnups proposed for NGNP (15 to 20% FIMA), the CO pressure in a  $\text{UO}_2$  fuel particle may be substantial, resulting in particle failure, especially under accident conditions. UCO was selected because the mixture of carbide and oxide components precludes free oxygen from being released due to fission. As a result, no carbon monoxide is generated during irradiation, and little kernel migration is expected. Yet, like  $\text{UO}_2$ , the oxycarbide fuel still ties up the lanthanide fission products as immobile oxides in the kernel, which gives the fuel added stability under accident conditions.

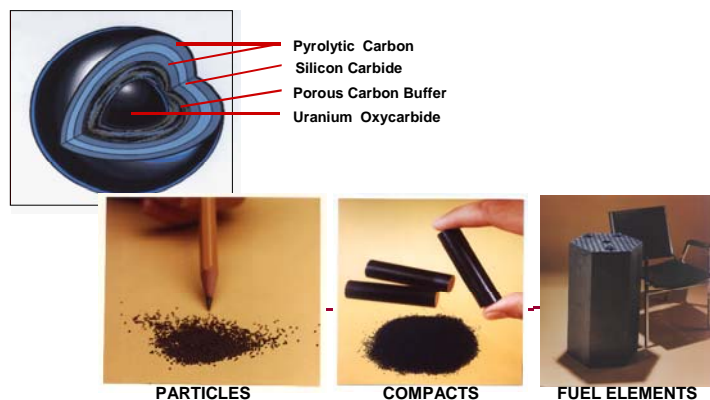


Figure A1.3. Cutaway of a TRISO-coated fuel particle and pictures of prismatic fueled high-temperature gas reactor fuel particles, compacts, and fuel elements.

For the pebble bed version of an NGNP, the coated particles are over-coated with a graphitic powder and binders. These over-coated particles are then mixed with additional graphitic powder and binders and then molded into a 50-mm-diameter sphere. An additional 5-mm fuel free zone layer is added to the sphere before isostatic pressing, machining, carbonization, and heat-treating.

Similarly, the prismatic version of the NGNP uses over-coated particles mixed with graphitic powder and binders to form a cylindrical compact about 50 mm long and 12.5 mm in diameter. After final heat treatment, these compacts are inserted into specified holes in the graphite blocks. Figure A1.3 shows a sketch of a TRISO-coated fuel particle and photographs of fuel particles, compacts, and fuel elements (prismatic blocks of graphite with fuel compacts and coolant channels) used in the high-temperature gas reactor at Fort St. Vrain. The AGR Fuel Development and Qualification Program is currently focusing on the prismatic fuel form.

### A1.3.1.2 History

Historically, U.S. high-temperature gas reactor fuel has experienced failures under irradiation while high quality German fuel did not. This fact is illustrated in Figure A1.4 where the krypton release to birth measurements from most of the U.S. and German TRISO coated fuel irradiation experiments are plotted versus fast fluence. The U.S. data from individual experiments is shown as lines whereas the yellow band in Figure A1.4 shows the range of the German data.

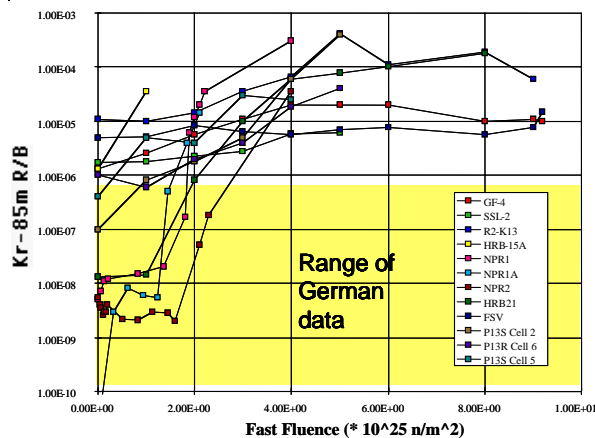


Figure A1.4. Krypton release to birth ratios versus fast fluence from a variety of U.S. and German fuel irradiation experiments showing the better performance of the German fuel.

The differences in the U.S. and German fuel performance have been traced to technical variations in the fabrication processes used in Germany and the United States. Three specific technical differences in the TRISO fuel coating layers produced by the respective fabrication processes have important impacts in terms of performance under irradiation and accident conditions: pyrocarbon anisotropy and density, IPyC/SiC interface structure, and SiC microstructure. A detailed review of the historical differences in the U.S. and German TRISO fuel fabrication processes and how they affect the AGR fuel forms development and manufacture can be found in the *Next Generation Nuclear Plant Research and Development Program Plan*, INEEL/EXT-05-02581, December 2004.

### A1.3.1.3 Fuel Manufacture

The fuel-manufacturing portion of the AGR program will produce coated-particle fuel that meets fuel performance specifications and includes process development for kernels, coatings, and compacting; quality control methods development; scale-up analyses; and process documentation needed for technology transfer. Fuel and material samples are produced for characterization, irradiation, and accident testing as necessary to meet the overall goals. Automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost will eventually be developed in later stages of the program and in conjunction with industrial partners.

The near-term activities focus on production of UCO kernels and coating of particles in a continuous process using a small (2-inch) laboratory-scale coater. The goal of these initial coating studies is to provide coatings produced under a range of coating conditions. The goal is to produce coatings like those produced by the German program in the late 1980s. However, coating variants are planned that will confirm our understanding of the historical coating fabrication database, and some of the variants will be irradiated in the first irradiation test, AGR-1. The second phase of the coating development involves scaleup of the continuous coating process to production size (e.g., 6-inch) coaters. The goal is to produce high quality coatings for performance demonstration and, ultimately, qualification.

Coated particles will then be over-coated and molded into cylindrical compacts using a matrix of graphite flour and carbonized resin. The thermosetting resin-based matrix and warm pressing compacting process selected for the program is similar to processes used in Germany and Japan, and it is a substantial departure from the thermoplastic matrix injection process used previously in the U.S. Development work is required to adapt the process to the U.S. fuel compact specifications.

In parallel with the fuel fabrication, fuel characterization will develop more advanced and robust techniques to measure key attributes of the fuel that can be integrated into a continuous production-scale coating process. Initial activities focus on developing improved anisotropy, coating layer thickness, and particle sphericity measurement techniques. Computer controlled sample positioning and digital imaging plus ORNL-developed image analysis software is used to quickly and easily analyze 1000's of particles, or particle cross-sections, for size and shape with a 1- to 2- $\mu\text{m}$  resolution. Example information from that system is shown in Figure A1.5.

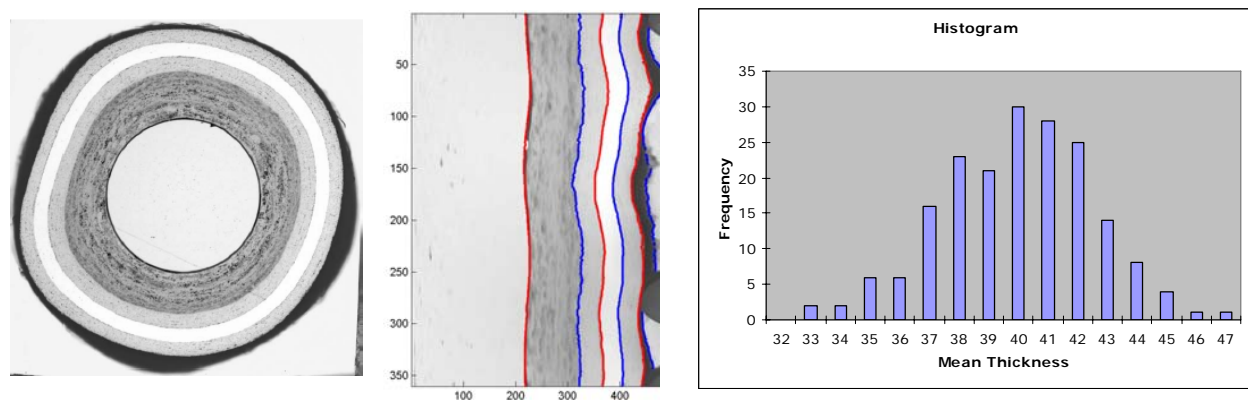


Figure A1.5. Example information from the ORNL computer automated optical characterization system. An IPyC histogram is shown on the right.

The crystallographic orientation in the pyrocarbon layers is measured by a scanning ellipsometry technique called the 2-MGEM (2-modulator generalized ellipsometry microscope). Figure A1.6 shows typical results from that equipment.

#### A1.3.1.4 Fuels and Materials Irradiations

The fuel and materials irradiation activities will produce fuel performance data to support fuel process development, to qualify fuel for normal operating conditions, and to support development and validation of fuel performance and fission product transport models and codes. The irradiations will also produce irradiated fuel and materials for post-irradiation examination and ex-core high-temperature furnace safety testing.

Eight irradiation capsules will be used to obtain the necessary data and sample materials. Details on each irradiation are listed in Table A1.2. The purpose of AGR-1 is to test a number of variants of fuel produced under different processing conditions from laboratory-scale coating equipment. AGR-2 will be a performance demonstration irradiation with fuel fabricated from a production-scale coater. Feedback to the fabrication process is expected following both AGR-1 and AGR-2. AGR-3 is devoted to obtaining data on fission gases and fission metals under normal irradiation conditions. AGR-4 will study fission product behavior in fuel compact matrix and graphite materials.

Given the very large number of fuel particles in a VHTR core, a large number of fuel specimens are needed to fully qualify the fuel and demonstrate compliance with the fuel failure specification. AGR-5 and AGR-6 are identical irradiations that will be used to qualify the fuel for the NGNP. AGR-7 and AGR-8 are irradiations designed to provide data with which to verify and validate fuel performance and fission product transport models.

Table A1.2. Planned AGR irradiation capsules.

Capsule	Task
AGR-1	Shakedown and early fuel
AGR-2	Performance test fuel
AGR-3	Fission product transport – 1
AGR-4	Fission product transport – 2
AGR-5	Fuel qualification – 1
AGR-6	Fuel qualification – 2
AGR-7	Fuel performance model validation
AGR-8	Fission product transport –3

The capsules will be irradiated in one of the large B positions at the Advanced Test Reactor at the Idaho National Laboratory. The large B position has a neutron spectrum very similar to that expected in a gas reactor. Preliminary calculations suggest that each capsule will be irradiated for 2.0 years to simulate a three to four year irradiation in the NGNP.

An important objective of the irradiation is to measure the fission gas release from the fuel and correlate it to the operating parameters in the irradiation. Each cell containing fuel specimens will be “sniffed” for fission gas. The sniffing gas will also be used to transport any fission gases released from the fuel to a location outside of the reactor where an ion chamber with enough sensitivity to indicate a single fuel particle failure (evident by a spike in its signal) will measure gross radiation in the line.

Data from the post-irradiation examinations will supplement the in-reactor measurements (primarily fission gas release-to-birth ratio measurements) as necessary to demonstrate compliance with the fuel performance requirements and support development and validation of the computer codes. This work will also support the fuel manufacture with feedback on the performance of kernels, coatings, and compacts. The various analyses and measurements that are proposed and their purpose are shown in the following Table A1.3.

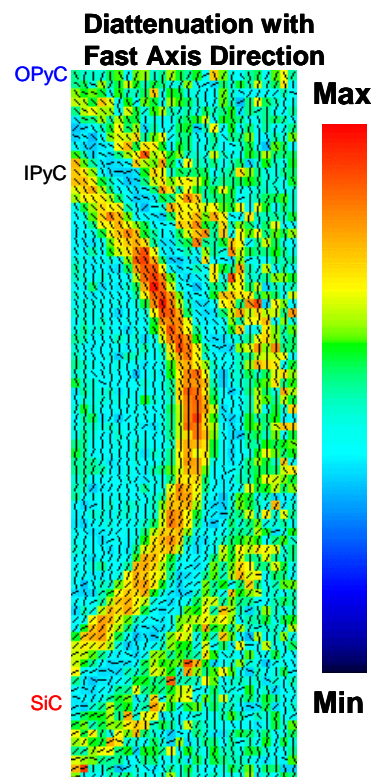


Figure A1.6. Typical results from the ORNL equipment for measuring pyrocarbon anisotropy.



Table A1.3. AGR post-irradiation examinations.

PIE Analysis or Test	Purpose
Gamma scan of the entire test train	Provides information to determine whether any fuel elements have broken or if a significant number of fission products have been released.
Post capsule disassembly specimen weights and dimensional measurements	Provides shrinkage or swelling characterization from irradiation
Optical metallography on cross sections of fuel pebbles or compacts	Provides physical characteristics of irradiated fuel particle coatings
Capsule components Gamma-scanning or leaching and gamma counting	Identifies migration and distribution of fission products following irradiation.
Electron microscopy of deconsolidated individual fuel particles	Looks for x-ray characteristics of specific fission products providing evidence of fission product accumulation at the IpyC/SiC interface, fission product attack of the SiC, and fission products outside the fuel particles.
Measurement of fuel particle failure fraction independently of the on-line release-to-birth ratio (R/B) measurements using the leach-burn-leach method	Provides a measurement of free uranium, which is converted to a SiC defect fraction
Irradiated microsphere gamma analysis (IMGA)	Provides a histogram of the ratio of Cs-137/Eu-152 based on all the particles in individual spheres compared to a normal distribution
Metallography following IMGA	Ties the microstructure of the anomalous particles to the fission product release.
Traditional burnup analysis	Determines the concentration of transuranics and minor actinides, to assess burnup

#### A1.3.1.5 Safety Testing

An important goal of this program is to evaluate the integrity and performance of the coated particle fuel under high-temperature accident conditions, which is essential to the safety case for the NGNP. In particular, three environments are of interest: helium, air, and steam. The irradiated TRISO fuel will be exposed to these environments for up to 500 hours. The exact composition of these environments are not known at present, but assumptions are that the test will be run at atmospheric pressure, and steam and air concentrations will be in the range of 10,000 ppm.

The data needed from safety testing are fission product release, TRISO coating layer integrity, and fission product distribution within fuel particles (corrosion likelihood) and fuel elements.

Post heating test activities include characterization of the TRISO coating layer integrity by optical metallography, including looking for evidence of SiC layer thinning and decomposition, chemical attack of the SiC, and the mechanical condition and microstructures of the SiC and PyC layers. Detailed test matrices will be developed as the program evolves.

#### A1.3.1.6 Fuel Performance Modeling

The high temperature gas reactor TRISO coated fuel performance computer codes and models will be further developed and validated as necessary to support the fuel fabrication process development and the NGNP design and licensing activities. The fuel performance modeling will address the structural,

thermal, and chemical processes that can lead to coated-particle failures. The models will also address the release of fission products from the fuel particle and the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. The new models are first-principles-based mechanistic, integrated, thermal-mechanical-physio-chemical-irradiation performance models for particle fuel, which have the proper dimensionality yet capture the statistical nature and loading of the fuel.

These models have had some success in predicting fuel failure mechanisms and rates in the U.S. fuel tested over the last decade and, thereby, facilitating a better understanding of TRISO coated fuel behavior. Such a tool can be very useful for both pretest and post-test predictions for any experiment performed in this program. Sensitivity studies with the model can also be used to identify critical materials properties data and constitutive relations whose uncertainty needs to be reduced because they drive the predicted performance of the coated fuel particle. Furthermore, use of piggyback cells in the irradiation capsules can be used to study those key individual phenomena in coated particles that have high uncertainty (e.g., shrinkage and swelling of the pyrocarbon, fission product release behavior in a purposely defective or initially failed particle). Moreover, some of the post-irradiation examination techniques can provide maps of fission products through the particle, which can be compared with model predictions of fission product transport through the coatings.

#### **A1.3.1.7 Fission Product Transport and Source Term**

Transport of fission products produced within the coated particles will be modeled to obtain a technical basis for source terms for advanced gas reactors under normal and accident conditions. The design methods (computer models) will be validated by experimental data as necessary to support plant design and licensing. The phenomena to be modeled include:

- Fission product release from the kernel
- Transport through failed coatings
- Deposition fraction of the released fission products in the compact or sphere matrix
- Deposition fraction of what gets through the compact on fuel element graphite (prismatic variant only)
- Deposition fraction of what gets out of the fuel element onto graphite dust and metallic surfaces in the primary circuit
- Re-entrainment of deposited fission products during an elevated temperature accident, or depressurization event
- Transport of fission products on dust particles, and subsequent release to the environment if the primary circuit is breached.

### **A1.3.2 Energy Conversion**

Planning for the Energy Conversion R&D has not yet been completed. Energy Conversion planning will be coordinated with industry participants.

## A1.3.3 Materials

### A1.3.3.1 Introduction

The NGNP Materials R&D Program will focus on testing and qualification of the key materials commonly used in VHTRs. The materials R&D program will address the materials needs for the NGNP reactor, power conversion unit, intermediate heat exchanger, and associated balance of plant. Materials for hydrogen production will be addressed by the DOE's NHI. The NGNP Materials Program Plan provides additional detail (see *NGNP Materials R&D Program Plan*, Sept 2004, INEEL/EXT-04-02347 Rev 1. The current organizational structure for management of the NGNP Materials R&D program is shown in Figure A1.7.

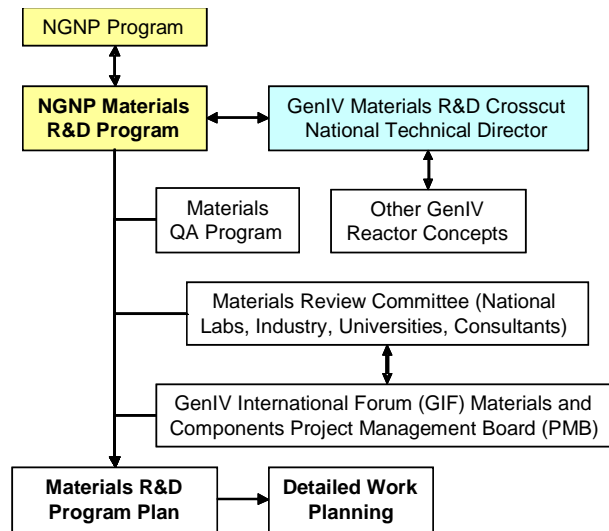


Figure A1.7. NGNP materials R&D organization.

### A1.3.3.2 Component Candidate Materials

A variety of materials options have been identified for potential use in the NGNP reactor and balance of plant components. This section summarizes the options currently identified by function.

Graphite will be the major structural component and nuclear moderator in the NGNP reactor core. The graphite used previously in the high-temperature gas reactor programs in the United States (designated H-451) is no longer in production, and thus replacement graphites must be found and qualified. Fortunately, likely potential candidates currently exist, including fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structures, fuel elements, and replaceable reactor components and near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components. These materials are expected to meet the requirements of the draft ASTM materials specification for the Nuclear Grade Graphite.

The reactor internals may include a core barrel, inside shroud, core support floor, upper core restraint, and shutdown cooling system shell and tubes. For the very-high-temperature components ( $>760^{\circ}\text{C}$ ), the most likely material candidates include variants or restricted chemistry versions of Alloys 617, X, XR, 230, 602CA, and variants of Alloy 800H. The upper limit of these materials, however, is judged to be  $1000^{\circ}\text{C}$ . Any component that could experience excursions above  $1000^{\circ}\text{C}$  would need greater very-high-temperature strength and corrosion resistance capabilities.  $\text{C}_f/\text{C}$  or  $\text{SiC}_f/\text{SiC}$  composites

are the leading choices for materials available in the near future for service that might experience temperature excursions up to 1200 °C. Compatibility of the metals with the helium coolant and irradiation resistance of the potential candidate materials needs to be addressed.

Several possible primary coolant pressure boundary systems are envisioned for the NGNP. These comprise (1) a large reactor pressure vessel containing the core and internals, (2) a second vessel containing an IHX and circulator (or a power conversion unit), and (3) a pressure containing cross-vessel joining the two vessels. Because these three vessels will be exposed to air on the outside and helium on the inside, emissivity of the material is an important factor regarding radiation of heat to the surrounding air to ensure adequate cooling. If the temperature can be maintained less than 375 °C by cooling or other means, conventional materials can be used. However, if the pressure boundary temperature is in the range of 375-500 °C, advanced materials will be required.

An IHX will be needed for hydrogen production and other process heat applications. It may also be desirable to use an indirect cycle for electricity production. The reactor coolant system pressure will be about 7 MPa. The pressure difference between the primary to secondary loop may be small (0.1 MPa) if helium is used for the intermediate heat transfer loop, or it may be large if a liquid such as molten salt is used. The leading IHX design for this cycle is a compact counter-flow configuration that involves channels passing through diffusion-bonded metallic plates. Transient thermal loadings could be a problem and will be addressed. Environmentally induced degradation of the metals from impurities in the helium or flow induced erosion is a concern. Aging effects are a concern for very long-time thermal exposure, since embrittlement could affect the performance of the IHX during thermal transients. Welding/brazing and fabrication issues exist. The leading potential candidates for service at temperatures of 900 to 1000 °C are Alloy 617, Alloy X, and Alloy XR. Other nickel-base alloys such as CCA617, Alloy 740, and Alloy 230 may be considered.

The key components of the NGNP power conversion unit will include turbines, generators, and various types of recuperators or heat exchangers. Considerable materials work may be involved in both the turbine and the generator components and existing component manufacturers are an excellent source of the needed materials information. The recuperator may be a modular counter-flow helium-to-helium heat exchanger, and current technology for the expected temperatures and pressures of operation is relatively mature.

Once appropriate materials have been designated for NGNP use, it will be necessary to gain ASME Boiler and Pressure Vessel (B&PV) Code acceptance of those materials at the desired operating conditions. To achieve B&PV Code acceptance, specific material information must be submitted to the appropriate subcommittees and will require significant justification. Once the material is accepted in Section II, it must also be submitted for construction approval in Section III, Subsection NH. While not strictly a part of the design methodology, the safety assessments required for licensing depend on much of the same materials and structures database.

#### **A1.3.3.3 Materials Qualification Testing Program**

The major elements of the NGNP Materials R&D Program are:

1. Test and qualify core graphite materials
2. Develop an improved high temperature design methodology for use with selected metals at very high temperatures
3. Develop ASME and ASTM codes and standards

4. Perform environmental testing and thermal aging of selected high temperature metals
5. Irradiation testing and qualification of reactor pressure vessel materials
6. Develop and qualify composites

7. Resolve reactor pressure vessel fabrication and transportation issues. The scope of the program is specifically designed to envelop the high-priority and long-lead materials R&D information anticipated to be required regardless of the NGNP system design chosen.

**A1.3.3.3.1 Graphite Testing and Qualification Project.** Significant quantities of graphite have been used in nuclear reactors and the general effects of neutron irradiation on graphite are reasonably well understood. However, models relating structure at the micro and macro level to irradiation behavior are not well developed.

The criteria for selecting graphites will consider whether the particular graphite can satisfy multiple reactor vendor design requirements, and whether there are sustainable precursors for extended production runs over the reactor's lifetime. A strategy for the selection process, acquisition process, and material receipt and storage requirements for the purchased graphite is being developed.

Engineers at the INL, in consultation with graphite experts at ORNL, have started an ATR creep capsule design. Prior Oak Ridge Research Reactor and Idaho Engineering Test Reactor graphite creep test capsule designs are being used as the basis for the new design. The graphite samples will be loaded under compressive stress and irradiated at representative temperatures. In addition to creep rate data, post-irradiation examination of the control samples will yield valuable irradiation effects data.

Mathematical models must be developed that describe and predict the behavior of nuclear graphite under neutron irradiation. Such models must be based on physically sound principles and reflect known structural and microstructural changes occurring in graphites during fast neutron irradiation, such as changes in crystallinity, pore shape, coefficient of thermal expansion (bulk and single crystal), etc.

Significant dimension and material property changes can occur in graphite subjected to neutron irradiation as illustrated in Figure A1.8 for the old H451 material. Therefore, a series of 36 NGB-10 nuclear graphite bend-bar samples were irradiated in rabbit capsules in the High-Flux Isotope Reactor (HFIR) at ORNL during FY-04. Each of the 18 rabbit capsules contained a SiC temperature monitor. PIE of the samples will begin at ORNL in FY-05.

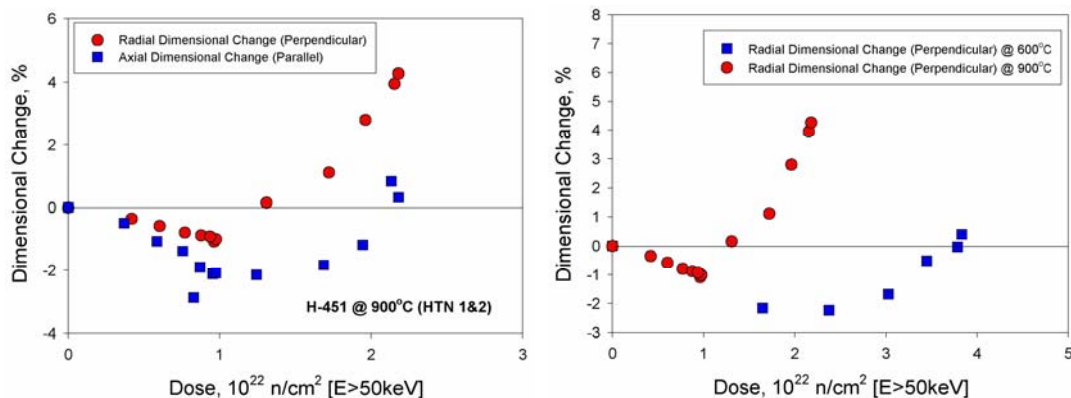


Figure A1.8. H-451 graphite dimensional changes as a function of orientation and temperature.

In addition, there is little data for the irradiation behavior of graphite at temperatures  $>1000^{\circ}\text{C}$ . Hence, a high-temperature graphite irradiation capsule for use in HFIR will be designed that will be capable of irradiating graphite samples at temperatures up to  $1200^{\circ}\text{C}$ .

**A1.3.3.3.2 High-Temperature Design Methodology Project.** The High-temperature Design Methodology (HTDM) project will develop the data and simplified models required by the ASME B&PV Code subcommittees to formulate time-dependent failure criteria that will ensure adequate life. This project will also develop the experimentally based constitutive models that will be the foundation of the inelastic design analyses specifically required by ASME B&PV Section III, Division I, Sub-section NH.

The HTDM project will produce test data, analyze results, and develop constitutive models for high-temperature alloys. Equations are needed to characterize the time-varying thermal and mechanical loadings of the design. Test data are needed to build the equations. The project will directly support the reactor designers on the implications of time-dependent failure modes and time and rate-dependent deformation behaviors. Safety assessments required by NRC will depend on time-dependent flaw growth, and the resulting leak rates from postulated pressure-boundary breaks. This requires a flaw assessment procedure capable of reliably predicting crack-induced failures and the size and growth of the resulting opening in the pressure boundary. Identifying an overall proven procedure is a part of this project. Figure A1.9 shows some of this work in progress.

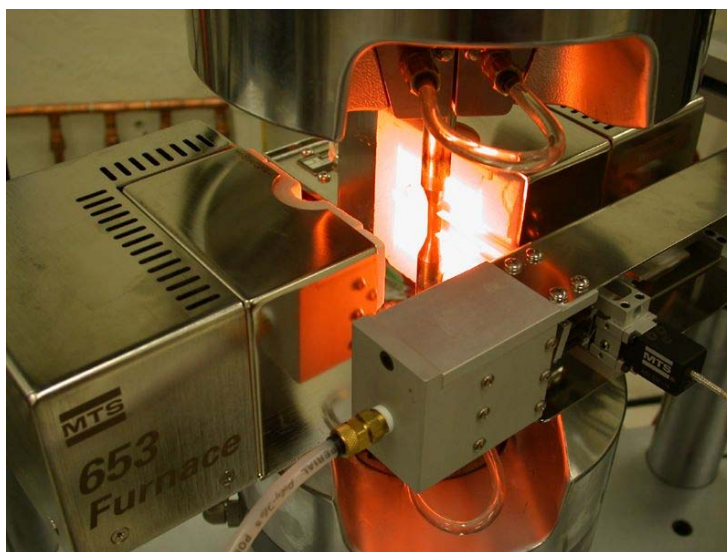


Figure A1.9. Creep-fatigue test in progress in air at  $1000^{\circ}\text{C}$  at the INL (furnace opened to show specimen).

**A1.3.3.3.3 Support for the ASTM and ASME Code.** There are a number of areas relating to ASTM standard method development and ASME B&PV Code development that must be pursued to meet the NGNP goals. Therefore, the NGNP Materials R&D Program must initiate a presence at ASTM and ASME B&PV Code meetings at the relevant committee and subcommittee level in order to incorporate new materials, and/or extend the application of materials presently in the Code, and/or further develop test standards.

Much of this effort will provide required technological support and recommendations to the Subgroup on Elevated Temperature Design (NH) as they develop methods for use of Alloy 617 at very high temperatures. In addition to inheriting the known shortcomings of Subsection NH, the Alloy 617 draft ASME Code case has a number of gaps and shortcomings that must be overcome before it can be

satisfactorily and reliably applied. Therefore, a new ASME Code case needs to be written. ASME design code development is also required for the graphite core support structures of the NGNP and later for the C/C composites structures of the core. A project team under Section III of ASME is currently undertaking these activities. Standard test methods for graphites and composites are also required to generate data that may be used in the design code. The ASTM DO2-F committee on Manufactured Carbons and Graphites is currently engaged in the final stages of developing a Standard Materials Specification for Nuclear Grade Graphite, and it is also developing several standard test methods for graphites (crystallinity by x-ray diffraction, surface area, thermal expansion, fracture toughness, and graphite oxidation, for example). The INL and ORNL will also support the formation of an ASTM working group on SiCf/SiC composite testing development, and ensure that guidelines for testing of tubular SiCf/SiC structures proceeds.

**A1.3.3.3.4 Environmental Testing and Thermal Aging Project.** The three primary factors that will most affect the properties of the metallic structural materials from which the NGNP components will be fabricated are (1) the effects of irradiation, (2) high-temperature exposure, and (3) interactions with the gaseous environment to which they are exposed. An extensive testing and evaluation program is needed to assess the effects of these factors on the properties of the potential materials to qualify them for the service conditions required.

Procedures for the evaluation of aged and “service-exposed” specimens will be developed. Properties evaluation will be performed on a limited number of materials, including Alloy 617, Alloy 800H, and Alloy X, that have been aged at temperatures as high as 870°C for long times in helium. It is expected that aging exposures of more materials will be performed to at least 25,000 h. Mechanical and microstructural properties of bulk and weld structures will be evaluated, and the determined experimental properties will also serve as input to and checks on the computational continuum damage modeling activity for predicting high-temperature life.

The out-gassing of nuclear grade graphites at very high temperatures may release significant impurities (H<sub>2</sub>O, CH<sub>4</sub>, CO<sub>2</sub>, CO, N, and H<sub>2</sub>) into the helium coolant of the NGNP. The overall stability of the NGNP helium environment must be evaluated to ensure that the testing proposed in various parts of the program is performed in environments having consistent chemical potentials relative to the expected environment. In addition, the corrosion of metals and nonmetals will be evaluated to establish baseline data where it does not exist. These tests will be performed at temperatures to include at least 50 °C above the proposed operating temperature.

**A1.3.3.3.5 Test and Qualify Reactor Pressure Vessel and Core Materials.** Some VHTR designs assume the use of higher alloy steel than currently used for LWR pressure vessels. The irradiation damage and property changes of these materials must be measured. Therefore, an irradiation facility that can accommodate a relatively large complement of mechanical test specimens will be designed and fabricated for placement in a material test reactor. This facility will replace the irradiation facility that was shut down last year at the Ford Test Reactor at the University of Michigan.

**A1.3.3.3.6 Composites R&D Project.** This program is directed at the development of C/C and SiC/SiC composites for use in selected very high temperature/very-high neutron fluence applications such as control rod cladding and guide tubes (30 dpa projected lifetime dose) where metallic alloy are not feasible. It is believed that SiC/SiC composites have the potential to achieve a 60-year lifetime under these conditions. The usable life of the C/C composites will be less, but their costs are also significantly less. The program will eventually include a cost comparison between periodic replacement of C/C materials and use of SiC/SiC composites.



Unlike monolithic materials, composites are engineered from two distinct materials using complicated vapor infiltration techniques. Therefore, the material properties may be affected when the component geometry or size is changed significantly. This is a major consideration, since small sample sizes and more suitable geometries are required for test samples. In addition, different composite architectures (i.e., weave angles, fiber tow counts, weave structures, etc.) can lead to differences in the engineered materials due to infiltration efficiency, fiber bending stresses, or matrix/fiber interface characteristics. The environmental conditions these materials will be subjected to may change the overall creep response of the composite (i.e., creep crack growth for fiber-reinforced materials). A typical Cf/C composite cross-section and weave pattern is shown in Figure A1.10.

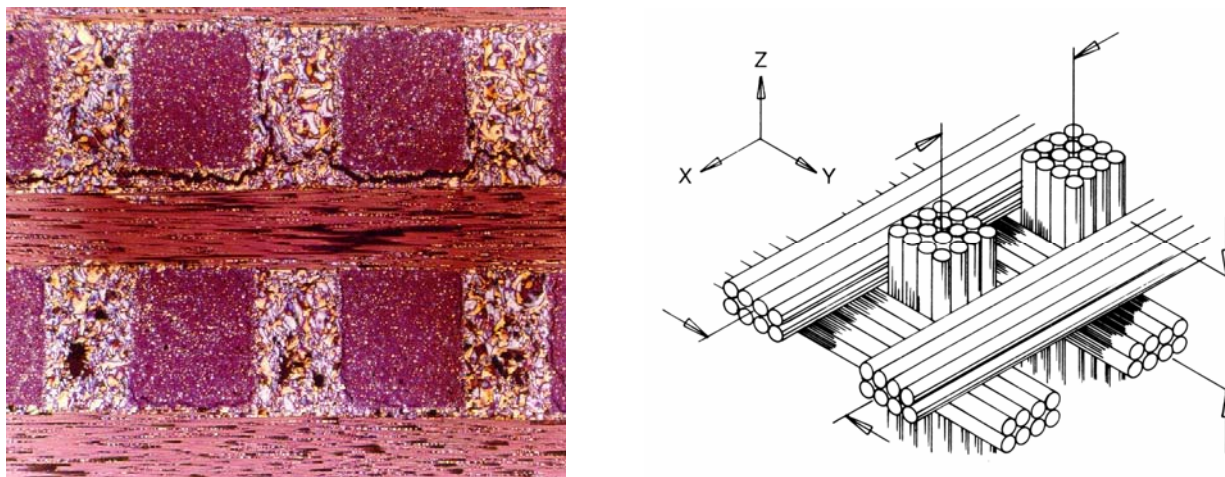


Figure A1.10. Typical Cf/C composite cross-section and weave pattern.

**A1.3.3.3.7 Data Management and Handbook.** The organizational structure for the preparation, control, etc., of NGNP data needs will be finalized for incorporation into the Gen IV Materials Handbook being developed in the Materials Crosscutting Program. Existing materials handbooks will be examined to determine what information might be extracted and incorporated into the Gen IV Materials Handbook. Once fully implemented, the Handbook will become the repository for the NGNP materials data and serve as a single source for researchers, designers, vendors, codes and standards bodies, and regulatory agencies. Near-term activities in this area will include assembling and inputting existing data on materials of interest to NGNP.

**A1.3.3.3.8 Additional Materials R&D.** Future materials R&D will also include the power conversion unit and generator; reactor pressure vessel emissivity; metallic reactor internals; intermediate heat exchanger and piping fabrication; hot duct liner and insulation; and valves, bearings and seals.

## A1.3.4 Design Methods and Evaluation

The NGNP Design Methods Development & Validation Program will develop the state-of-the-art software analysis tools, and supporting data, required to calculate the behavior of the NGNP system during normal and off-normal scenarios. The software tools discussed here include those necessary to calculate the neutronic behavior, the thermal-hydraulic behavior, the interactions between neutronics and thermal-hydraulics, and structural behavior where necessary. The NGNP Methods Program is designed to be interactive across the appropriate parts of the DOE complex as well as with other university and industrial nuclear community stakeholders, and it will include their feedback through a peer-review process.



The Design Methods Development and Validation R&D implementation methodology shown in Figure A1.11 is as follows:

1. Selection of the most challenging scenarios together with the dominant phenomena in each, by utilizing a PIRT
2. Internal validation of the software tools and data required to calculate the NGNP behavior in each scenario
3. External validation of the software tools via non-NGNP Project nuclear engineering community participation in international standard problems
4. R&D performed through GIF-member & NGNP Project collaborations centered in International Nuclear Engineering Research Initiatives
5. R&D performed through university and NGNP Project collaborations centered in Nuclear Engineering Research Initiatives or GIF Project Management Board agreements
6. Software development, when validation findings show that certain models are inadequate
7. Analysis of the operational and accident scenarios, and finally
8. Review of the global process, and the process ingredients, using experts outside the program.

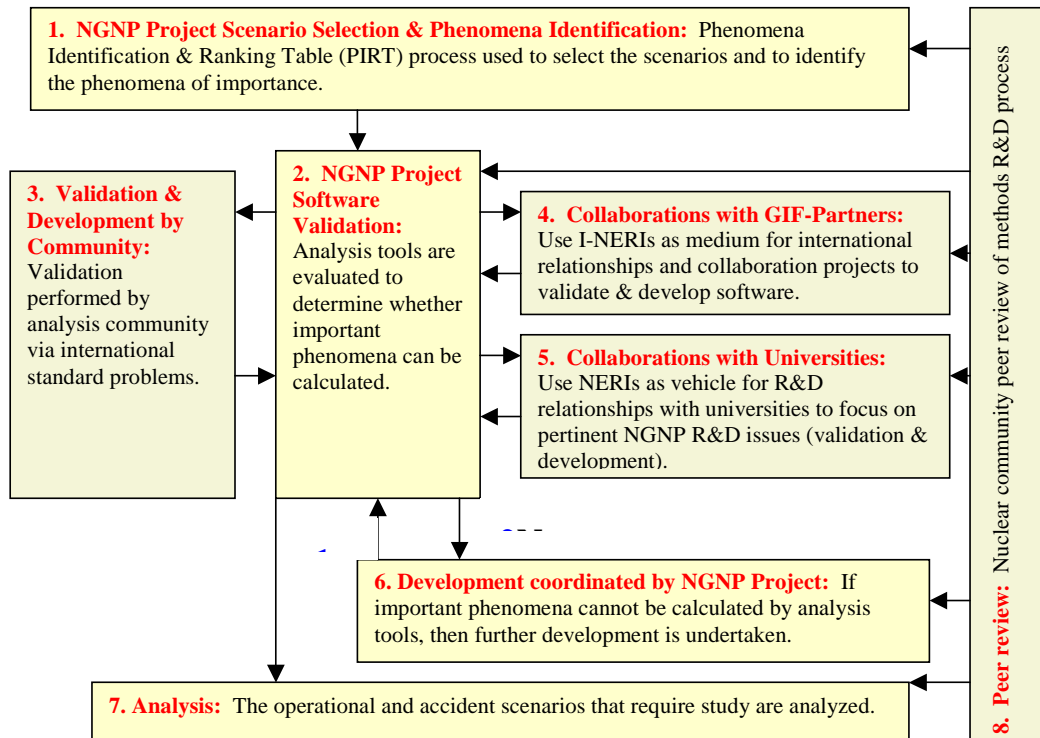


Figure A1.11. Methods R&D process.

A rigorous PIRT analysis of the NGNP has not been performed since the design has not yet been identified. However, based on the accumulated knowledge of the advanced gas-cooled reactor vendor

community and engineering judgment, a “first-cut” PIRT has been defined and used to specify the FY-05 and subsequent years R&D. Once a reactor design is specified, the design methods development and validation R&D requirements will be aligned with the design. The PIRT has identified a number of important phenomena (see *Next Generation Nuclear Plant Research and Development Program Plan*, INEEL/EXT-05-02581, December 2004). Based on these phenomena, the R&D for FY-05 R&D and beyond will be focused around five major tasks:

1. Computational fluid dynamics (CFD) code validation experiments: lower plenum, hot channel, and reactor cavity cooling
2. Validation of thermal-hydraulic software, e.g., CFD calculations of exit fluid temperature from hot-channel and lower plenum turbulence; core analysis methods development
3. Core physics methods development
4. Nuclear data tasks
5. Liquid salt-cooled methods development and design assessments

#### **A1.3.4.1 Validation of Thermal Hydraulics Software and CFD Codes Including CFD Validation Experiments**

The thermal-hydraulics of the NGNP encompasses the heat generation by the fuel; its transport to the helium coolant; and the laminar, transition, or turbulent flow of the helium as it flows from the upper plenum through the core, into the lower plenum, then out the exit duct to the intermediate heat exchanger or power generation vessel. Also included are the heat losses from the reactor vessel during normal operation as well as accident scenarios that may occur from failures in the system. The system designed to remove the heat in the event of an accident, the reactor cavity cooling system, is also included in the thermal-hydraulics of the NGNP.

Advanced simulation tools are available to simulate turbulent flow and heat transfer in complex systems. These tools must be validated for application on the NGNP. CFD codes are needed to simulate regions of complex turbulent flow in the plant. Because of the size and complexity of the plant, thermal-hydraulics systems analysis codes will also be applied, in conjunction with CFD codes, to analyze the plant.

The high-priority research areas identified in the “first-cut” PIRT include (1) the core heat transfer, (2) mixing in the upper plenum, the lower plenum, hot duct, and turbine inlet, (3) the heat transfer in the reactor cavity cooling system, (4) air ingress following a system depressurization, and (5) the behavior of the integral system during the key scenarios, including the contributions of the balance-of-plant. These R&D areas are outlined in Table A1.4 together with a summary of the key needs.

**A1.3.4.1.1 CFD Code Validation Experiments.** The experiments that stem from the areas identified in Table A1.4 are discussed at a summary level below. More detailed discussion can be found in the *Next Generation Nuclear Plant Research and Development Program Plan*, INEEL/EXT-05-02581, December 2004. Some potential issues identified to date include “hot streaking” in the lower plenum evolving from “hot channels” in the core, the geometric transition from the lower plenum into the outlet duct and the resulting temperature distribution in the short outlet duct, “hot plumes” in the upper plenum during “pressurized cooldown” (loss of flow accident), and parallel flow instability in the core during pressurized cooldown [Bankston 1965; Reshotko 1967]. Several of these phenomena are pertinent to pebble-bed versions of the NGNP as well as the block versions. The initial studies will concentrate on the

coolant flow distribution through reactor core channels (hot channel issue) and mixing of hot jets in the reactor core lower plenum (hot streaking issue), phenomena that are important both in normal operation and in accident scenarios.

Table A1.4. Methods Thermal-Hydraulics R&D Areas

<b>R&amp;D Area</b>	<b>Related R&amp;D</b>	<b>Study Area</b>	<b>Need</b>
1. Core Heat Transfer	Mixed convection experiment, heated experiments, core heat transfer modeling, bypass experiments, system performance enhancements, Sana experiments.	Experimental (E), CFD, & systems analysis codes (S)	The core heat transfer, both with cooling flow (operational conditions) and without cooling flow (DCC and PCC), are instrumental in setting the maximum temperature levels for fuel and material R&D (core graphite, structural materials, and heat load to RCCS). The core heat transfer will determine the material selection and configuration in the NGNP core, vessel, and RCCS designs.
2. Upper & Lower Plenums (UP & LP)	HTTR UP & LP, HTR-10 UP & LP, MIR, heated experiments, scaled vessel, jets & cross-flow data, upper plenum experiments, system performance enhancements.	E & CFD	<p>Circulation in the upper plenum is important during the PCC scenario since hot plumes rising from the hot core may impinge on the upper head structures and lead to a potential overheating of localized regions in the upper vessel.</p> <p>The degree of lower plenum mixing determines both the temperature variations and the maximum temperatures that are experienced by the turbine blades, the lower plenum, hot duct, and power generation vessel structural components. The lower plenum mixing will determine the material selection and configuration in the NGNP lower plenum, hot duct, power generation vessel, and turbine designs.</p>
3. RCCS	ANL (air-cooled), Seoul National University (water-cooled), HTTR RCCS, fission product transport, system performance enhancements	E, CFD, & S	The heat transfer efficiency of the RCCS will determine the overall design concept (whether air-cooled is sufficient or water-cooled is required in accordance with either a confinement or containment RCCS design), plus material selection of outer vessel wall, coatings (e. g., selection of materials with emissivities that change with surface temperature), natural circulation characteristics, etc.
4. Air Ingress	Diffusion model development, NACOK experiment	E, CFD, & S	A Generation IV reactor system should be able to survive the most challenging accident scenarios with minimal damage and thus should be able to resume operation in a minimum time frame. The system must be shown to sustain minimal damage following potential air ingress into the core region.
5. Integral System Behavior	HTTR, HTR-10, AVR, fission product transport, CFD and systems analysis code coupled calculations, behavior of balance-of-plant components (IHX, turbine, compressor, reheater), analyses of pre-conceptual design, conceptual design, preliminary design, and final design	E, CFD, & S	The ultimate system characterization, to show the final design is capable of meeting all operational expectations and of surviving the most challenging accident conditions, is performed using validated software tools. The tools consist of the neutronics and thermal-hydraulics software (coupled CFD and systems analysis software) used in concert. This step is the culmination of the comprehensive R&D effort outlined herein.

The general approach is to develop benchmark experiments needed for assessment in parallel with CFD and coupled CFD/systems code calculations for the same geometry. In each case, the benchmark experiments must be linked to the “potential” design by comprehensive scaling analyses that illustrate the relationships between the experiments and design—to ensure the experiments yield benchmark data that are within the design’s operational or postulated accident envelope.

Velocity and turbulence fields will be measured in the INL's unique Matched-Index-of-Refraction (MIR) flow system (see Figure A.1.12); these data will be used to assess the capabilities of the CFD codes and their turbulence models and to provide guidance in improving the models. Heat transfer experiments will be developed and accomplished for the same purposes. Existing databases from experiments, direct numerical simulations, and large eddy simulations will also be utilized where appropriate.

The following experiments aim to produce data to validate the NGNP CFD software.

1. Core heat transfer experiments:
  - a. Turbulence and stability data from vertical cooling channels (2005-2007)—experiments defined to study (a) the natural circulation-driven jets that emerge from the hottest channels and that may impinge on the vessel upper head structures during a pressurized conduction cool down scenario and (b) the factors that influence the hottest coolant exit temperatures during normal operation.
  - b. Bypass flow studies (2007-2008)—experiments designed to characterize the variation in the bypass flow during the lifetime of the NGNP as well as the influence of the bypass flow on other variables of importance, e.g., cooling flow, fuel temperatures, etc.
  - c. Exit flows in pebble beds (2006-2008)—the factors that influence the exit flow from the pebble bed into the downstream plenum are characterized and correlated to define their effect on downstream power conversion equipment.
2. Upper & lower plenum fluid behavior experiments:
  - a. Fluid dynamics of lower plenums (2005-2008)—a series of isothermal experiments designed to study the core coolant jet interactions and mixing as a function of location and flow conditions (flow rates and wall interactions) using the MIR facility shown in Figure A1.12.
  - b. Heated flows in lower plenums (2005-2007)—an experimental series that complements item 2.a by measuring the influence of the buoyant contributions when exit jet temperatures are considered.
  - c. Interactions between hot plumes in an upper plenum and parallel flow instabilities (2007-2009)—validation data are measured to characterize the coolant jet plumes, and their interactions, in the upper plenum during a pressurized conduction cooldown scenario. The plumes from the hottest channels will impinge on the upper plenum structural members.

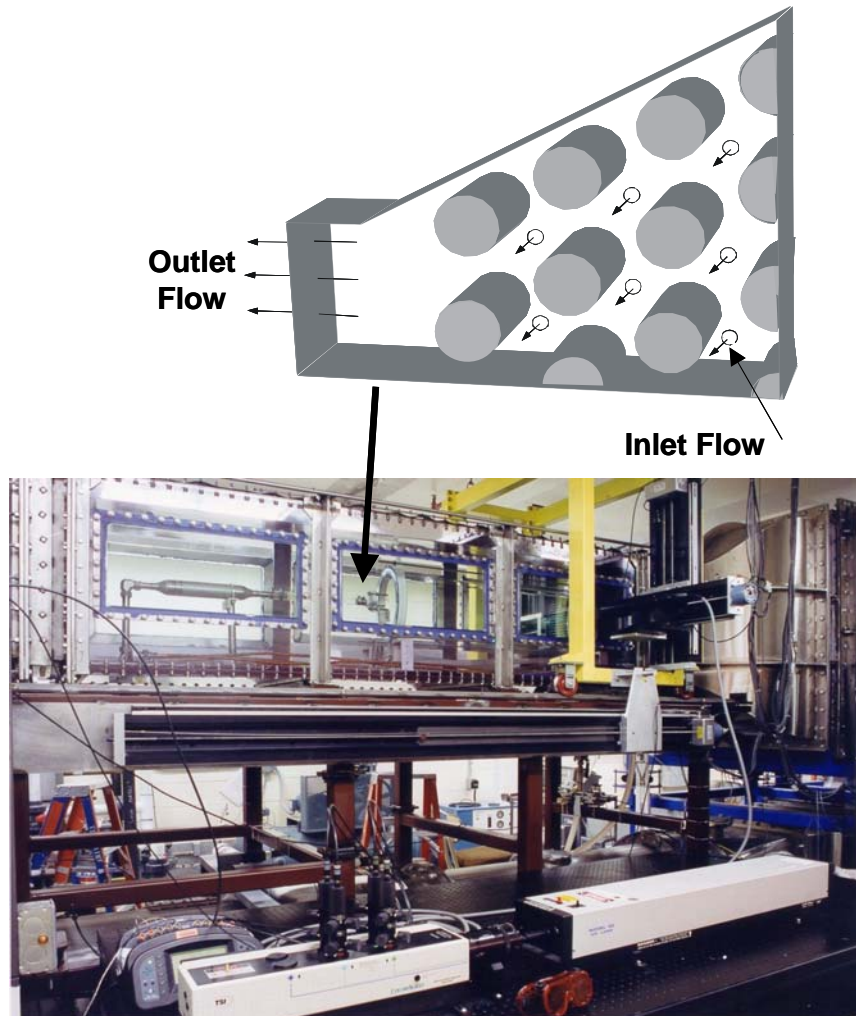


Figure A1.12. Matched-Index-of-Refractive flow system and a conceptual model design to study important flow features in a VHTR lower plenum.

3. Air ingress experiments: heat transfer and pressure drop of mixtures of air and helium (2006-2007)—intended to supplement existing data (e.g., International Atomic Energy Agency benchmark data) by recording measurements specific to the intended NGNP design; the experiments provide scaled data for the depressurized conduction cooldown scenario when air component gases move into the lower plenum and hence the core by diffusion.
4. Larger scale vessel experiments: to examine the behavior in the core, in the plenums, and the interactions between them (2010-2014)—a series of experiments performed in a scaled mockup of the near-final or final NGNP design to characterize the flow behavior specific to the defined geometry, e.g., characteristic turbulent behavior stemming from the area expansions, contractions, and changes in direction.
5. Integral Experiments: HTTR and HTR-10 test reactors—the only integral advanced gas-cooled experimental facilities in the world (prismatic and pebble-bed respectively) will be used to obtain representative data that includes various phenomena interactions throughout the reactor system to enable validation of the phenomena interaction calculations to be performed.

6. Reactor Cavity Cooling System Experiments—data representative of the behavior of this key heat removal system will be measured for validation purposes to characterize the system behavior during operational conditions, depressurized conduction cooldown conditions, and pressurized conduction cooldown conditions. These data are crucial in validating the models that directly influence the fuel temperatures both at steady state and accident conditions.

In general, two types of experiments are planned: fluid dynamics measurements and heated flow studies. The purpose of the fluid dynamics experiments is to develop benchmark databases for the validation of CFD solutions of the momentum equations, the scalar mixing, and the turbulence models for typical NGNP geometries in the limiting case of negligible buoyancy and constant fluid properties—that is, when the flow is turbulent and momentum-dominated. The intent of the heated flow experiments is to provide data on the modifications of the thermal hydraulic behavior (and proposed turbulence models) as additional effects, such as gas property variation and buoyancy, become important.

#### ***A1.3.4.1.2 Thermal-Hydraulic Design Methods Development, Validation, and Analysis.***

The modeling strategy chosen for this effort is to make use of both thermal-hydraulic systems analysis codes and CFD codes. The reference codes chosen are the RELAP5-3D systems code and the Fluent and STAR-CD CFD codes. However, other codes, such as GRSAC, Abaqus, and NPHASE will be used to supplement the reference codes. A systems analysis code is needed to model the integrated behavior of the entire NGNP system, including the interactive coupling of the reactor with the hydrogen and power producing components (such as, the intermediate heat exchanger, turbine, compressor, reheaters, etc). CFD software is needed to analyze the fluid behavior wherever two- or three-dimensional fluid behavior is expected, particularly in plenums and cavities. Regions of applicability for the NGNP include the upper and lower plenums, the hot duct, the intermediate heat exchanger or turbine inlet region, as well as the reactor cavity cooling system.

Two commercial CFD reference codes (Fluent and STAR-CD) are presently being used, and a university-developed or a national laboratory-developed CFD code may also be used. However, it is suspected that none of them will meet all of the NGNP analysis requirements, and thus some modifications will be required. Consequently, a three-track approach will be used to meet the CFD analysis needs for the NGNP:

Track 1: validation of currently available CFD software,

Track 2: modification of existing tools as necessary, and

Track 3: pursuit of R&D to obtain more efficient and effective simulation tools that may take several years to mature.

The near-term thermal-hydraulics tasks follow the first track: validating existing tools. As the CFD tools are validated, it may become necessary to add new turbulence models or pursue other modeling strategies, such as large eddy simulation (LES) or Direct Numerical Simulation (DNS), thus following Track 2. Track 3 is designed to ensure that needed simulation tools will be available in the future that are more efficient and capable than existing tools.

The key technical issues identified by the first-cut PIRT, as summarized in Table A1.4, are the basis for defining the code development, validation, and analysis R&D program. The R&D activities, organized based on the five key technical issues, are listed below:

1. Core heat transfer model validation, development, and analysis: (a) Convective Heat Transfer during Normal Operation – CFD analysis and systems analysis calculations in (FY-05) and

experimental data to be used for validation (FY-06 & FY-07), (b) Convective Heat Transfer during PCC: (FY-05 to FY-08), (c) Axial and Radial Conduction during PCC and DCC: (FY-06 to FY-08), and (d) Core bypass

2. Lower and upper plenum coolant flow mixing validations, development, and analysis: (a) Lower plenum (CFD model validation from FY-05 to FY-08) and (b) Upper Plenum flow (validation of coupled RELAP5-3D/Fluent model) (FY-09)—see Figure A1.13.

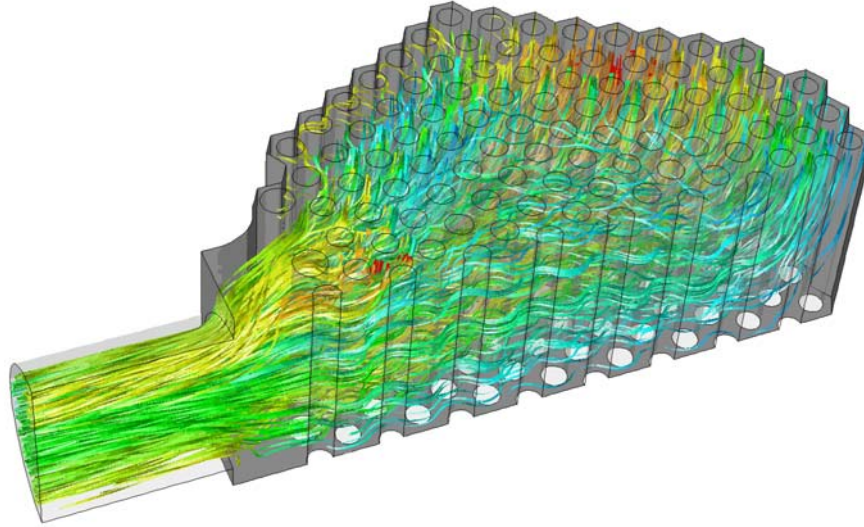


Figure A1.13. Preliminary CFD calculation of mixing in lower plenum (courtesy of Fluent, Inc).

3. Reactor Cavity Cooling System validation, development, and analysis (FY-06 to FY-09)
4. Air ingress and Fission Product Release validations, development, and analysis including (a) Air ingress scenarios calculations and RELAP5-3D multi-species model extension (FY-06) and (b) Linking of RELAP5-3D to coupled models PARFUME and VICTORIA (FY-10)
5. Integral system behavior validations, development, and analysis (FY-07 to FY-09)

#### **A1.3.4.2 Reactor Kinetics and Neutronics Analysis Development**

The design and operational analyses of the NGNP require the ability to carry out the following reactor physics computations: (1) cross section preparation and fuel assembly spectrum calculations to produce effective nuclear parameters for subsequent global reactor analysis, (2) static reactor analysis for core design and fuel management, (3) reactor kinetics and safety analysis, (4) material-neutronics interface, and (5) validation and verification. This section focuses on the development of a suite of deterministic code systems, including spectrum codes, a lattice physics code, and nodal diffusion codes, that can be used for efficient and accurate design of the NGNP. In order to accomplish the project goal efficiently, existing codes will be used as the basis of the new code suite with the addition of required functionalities for VHTR applications.

The INL and ANL will lead the research efforts for these tasks and are cooperating on the identification and development of a code suite that incorporates the techniques required for accurate analyses of all current candidate NGNP concepts. It is anticipated that the bulk of the code development



effort will be completed in the first five years of the overall effort, FY-05 through FY-09. After that, code maintenance, validation, and application to the ongoing NGNP design effort will continue for the duration of the project.

**A1.3.4.2.1 Fuel Cell and Assembly Spectrum and Cross Section Generation.** For NGNP applications, both the DIF3D/REBUS system and PEBBED require cross-section data preparation using specialized techniques that are not implemented in current software in the required form. Current cross sections used by PEBBED and DIF3D are calculated externally and passed to these global reactor analysis codes as input. For PEBBED simulations (and DIF3D analyses of the New Production Reactor), these cross sections have been calculated by the INL's COMBINE code or by MICROX-2, both of which need enhancements to properly handle certain phenomena that are characteristic of graphite-moderated reactors with highly heterogeneous fuel arrangements. The following subtasks have been identified under this task:

- Development of a method for improved treatment of double heterogeneity using improved Dancoff Factors.
- Develop the interface between a spectrum code and a pebble bed reactor core simulator. Cross section and core simulation calculations must be executed simultaneously and iteratively to obtain the proper burnup conditions in each spectral zone (the pebble bed reactor analog to an assembly or block).
- Identify or develop an assembly code for prismatic block cross-section generation. The lattice transport code, used for group constant generation, must be able to treat the double heterogeneity properly and account for spectral variations across the basic lattice unit via appropriate neutron transport computations.

**A1.3.4.2.2 Static Analysis for Evaluations of Criticality and Power Distribution.** The fundamental quantity in reactor physics analysis, determining all other aspects of core behavior, is the neutron flux. Extremely accurate calculations of the neutron flux, accounting for great geometric detail, can be made with Monte Carlo codes such as MCNP. However, Monte Carlo codes are still prohibitively expensive for use in iterative design calculations involving evaluation of local power distributions and small reactivity effects. Nor can current coupled Monte-Carlo-depletion codes be applied to the pebble bed reactor (Figure A1.14). This section focuses on development of deterministic codes for use on pebble bed static reactor core design and fuel optimization. It includes work in the following areas:

- Fuel management and design optimization: PEBBED possesses an advanced optimization routine (a genetic algorithm) that allows automated searches for optimal core designs and fuel loading patterns. Such methods have not yet been developed for the prismatic reactor and will be developed. Core optimization and fuel loading will also be automated to produce viable cores within practical time limits. One or more advanced optimization approaches will be explored and implemented in DIF3D/REBUS.
- Neutron Transport: The pebble bed reactor contains a large gas plenum above the core through which pebbles are dropped. A transport calculation will be developed to accurately predict neutron transport in these regions. Also, a NERI project led by F. Rahnema, of the Georgia Institute of Technology, is investigating neutron transport techniques accurate treatment of gas plena and control rod regions in the pebble bed reactor. The DeCART, Attila, and EVENT codes will be explored for use in 3D deterministic transport calculations for some applications, such as accurate gamma transport and deep shielding.

- Isotope Depletion: The DIF3D/REBUS-3 code system is capable of multi-group flux and depletion calculations in hexagonal-Z geometry, which will be adapted to prismatic NGNP reactor problems with limited effort compared to other codes.
- Pre-asymptotic core analysis in the pebble bed reactor: A theoretical formulation for pre-asymptotic core analysis is under way at the INL and will ultimately lead to time-dependent solutions of the coupled pebble-flow/burnup problem.
- Non-axial pebble flow in the pebble bed reactor: A method and code will be developed that link together depletion zones along the true flow path of pebbles in a pebble bed reactor, which is not strictly axial, particularly near the discharge tubes.

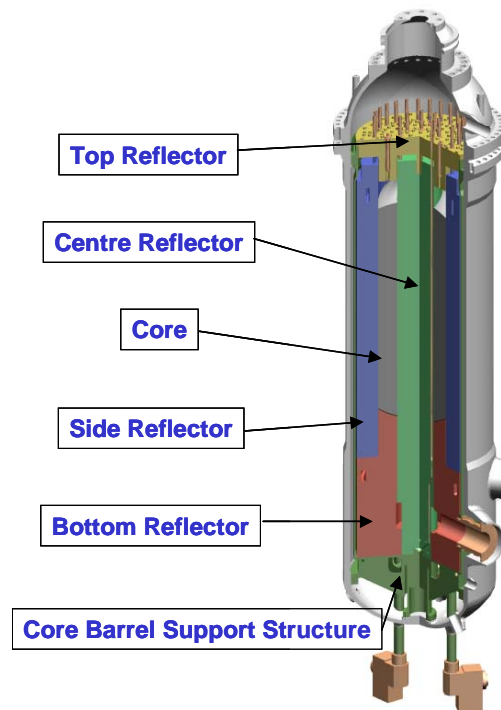


Figure A1.14. Sketch of the PBMR.

#### **A1.3.4.2.3 Kinetics, Thermal Module**

**Coupling, and Feedback.** High-fidelity kinetics methods are important for core transients involving significant variations of the flux shape, but these methods have not been systematically applied to graphite-moderated, helium-cooled reactors. In the future, integrated thermal-hydraulics and neutronics methods will be extended to enable modeling of a wider range of transients pertinent to the NGNP. Required advances include increasing the efficiency of the coupling approaches and improving the representation of cross section variations.

A cooperative research effort between Penn State University and PBMR (Pty), Ltd of South Africa is underway to develop a coupled neutronics-thermal-hydraulics code for pebble-bed reactor transient and safety analysis starting with the NEM code. The INL has been invited to participate; the PEBBED code would be used to generate steady-state conditions to be fed to the transient code. This is a good bridge to a complete steady state and transient code.

Nodal diffusion and transport kinetics capabilities have been developed for the DIF3D code. These capabilities have been successfully applied for transient analysis of thermal reactor systems by integrating them in a system analysis code, SASSYS. Initial estimates indicated that a multi-group analysis (about 20 groups) is required to represent accurately the reactivity effect of spectral change. The multi-group capability of DIF3D would be attractive for integration with a system code, such as RELAP5/ATHENA, that can be utilized for the analysis of the NGNP.

**A1.3.4.2.4 Material-Neutronics Interface.** Of particular importance is the change in material properties caused by radiation. For example, the thermal conductivity of graphite is significantly degraded as radiation damage accumulates. Similarly, some nuclear properties, such as the scattering cross sections, are altered by the damage. During transients, the increase in temperature may anneal some or all of the damage, resulting in (partial) property recovery. This could imply, for example, that the scattering cross section would increase during a transient, resulting in stronger thermalization properties and an increase in reactivity. Other similar phenomena are believed to occur that also have a potential

impact on the safety of the NGNP during extreme transients. The feedback mechanisms just described must be incorporated into the kinetics codes.

**A1.3.4.2.5 Validation, Verification, and Ongoing Improvement of Code Suite.** The resulting suite of deterministic codes developed above will be verified against Monte Carlo and deterministic codes and against integral experiments. The double heterogeneity treatment will be examined for detailed fuel block and pebble problems by comparing the lattice code solutions with continuous-energy Monte Carlo solutions. The whole-core solution scheme will be verified against multi-group Monte Carlo solutions using pre-calculated multi-group cross sections and homogenized fuel-element models. The pebble bed reactor solution will also be compared against results from the code VSOP [Teuchert et al. 1980].

#### **A1.3.4.3 Nuclear Data Measurements, Integral Evaluations, & Sensitivity Studies**

Accurate differential nuclear data libraries and well-characterized and accurate integral benchmark information are required for all computational reactor physics tasks associated with NGNP design and operation. Differential nuclear cross section data for all materials used in the reactor are required as input to the physics codes. Furthermore, integral benchmark experiment data for relevant existing critical configurations are required for physics code validation. Finally, rigorous sensitivity studies for representative NGNP core designs are required for prioritizing data needs and for guiding new experimental work in both the differential and integral regimes. The above needs will be satisfied through the following activities discussed below.

**A1.3.4.3.1 Sensitivity Studies.** ANL in collaboration with INL will quantify uncertainties in computed core physics parameters that result from propagation of uncertainties in the underlying nuclear data used in the various modeling codes. This study will aid in further quantifying the need for additional cross-section measurements and provide a guide in planning of future measurements and evaluations. Formal sensitivity and uncertainty analysis will be performed to identify the nuclides that contribute to calculational uncertainties and quantify the propagated uncertainties in the context of the currently anticipated NGNP core designs. The NGNP gas-cooled prismatic core design will be the basis for this initial study [MacDonald et al. 2003]. Subsequent studies will encompass the other candidate concepts. Sensitivity coefficients will be calculated by generalized perturbation theory codes and folded with multi-group covariance data (where available) to derive propagated uncertainties in computed integral reactor parameters arising from the nuclear data. Parameters to be evaluated include reactivity, peak power, reaction rate ratios, nuclide inventory, safety coefficients, etc. Most of the effort will be conducted during the first three years, FY-05 through FY-07.

**A1.3.4.3.2 Integral Nuclear Data Evaluations.** The computer codes used in NGNP design and safety analyses must be benchmarked against appropriate experimental data. During FY-04, under DOE Generation IV crosscut funding, the INL and ANL studied various experimental and prototypical HTGRs developed since the 1960's to assess their potential as benchmarks.

The HTR-10 test reactor (Figure A1.15) was chosen for the first pebble bed reactor benchmark that will undergo full evaluation, and HTTR and VHTRC were chosen for the first block-reactor benchmarks. An additional assessment exercise will include any available liquid salt cooled concepts that may be suitable as benchmarks. Current plans are to conduct this assessment, as a collaborative effort of INL, ANL, and ORNL, in FY-06 or FY-07.

The next steps involve evaluation and documentation of the identified facilities to provide benchmark specifications accepted for validation of physics modeling codes. The work will be conducted under the International Reactor Physics Evaluation Project (IRPhEP), an international effort endorsed by

the Organization of Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC) in June of 2003. The INL, which provides leadership for the IRPhEP Technical Review Group, and ANL will contribute NGNP-specific benchmarks evaluated under this R&D Plan.

Initially, the INL will evaluate the HTR-10 test reactor and possibly PROTEUS. ANL will evaluate either HTTR or VHTRC. Other appropriate benchmarks will be evaluated in later stages of the overall integral benchmark effort encompassed by this Plan. It is anticipated that evaluations for most, if not all, of the higher priority NGNP-specific gas-cooled facilities identified in the assessment will be completed during the first five years, FY 2005 through FY 2009.

#### **A1.3.4.3.3 Differential Nuclear Data Measurements.**

The current NGNP design will have a harder thermal neutron spectrum and two to three times the burnup of a light water reactor. As a result, improved cross section measurements are needed in certain neutron energy ranges for some isotopes, in particular  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ , and  $^{242}\text{Pu}$ .

The INL, in partnership with ANL and various university and international collaborators, will conduct a research program of measurements for the actinides of interest at the ANL Intense Pulsed Neutron Source (IPNS). In the first five years, FY-05 through FY-09, the differential data measurement campaign will be focused on development of data for  $^{240}\text{Pu}$  and  $^{242}\text{Pu}$ , followed by measurements for  $^{241}\text{Pu}$ . After this initial campaign is complete and the data published, attention will turn to the higher actinides with specific priorities set by the cognizant international nuclear data working groups.

#### **A1.3.4.4 Liquid Salt-Cooled NGNP Methods Development and Design Assessments**

##### **A1.3.4.4.1 Development of Thermal-Hydraulics Methods for Liquid Salt-Cooled NGNP Design.**

Two liquid salt coolants will be implemented into the RELAP5-3D code to support the pre-conceptual design of the liquid salt-cooled NGNP. The salts currently being considered are  $\text{Li}_2\text{BeF}_4$  (flibe) and  $\text{LiFNaFKF}$  (flinak). This work will be done in collaboration with similar work at ORNL.

##### **A1.3.4.4.2 Liquid Salt-Cooled NGNP Thermal-Hydraulic and Neutronics Assessments.**

During FY-05, ORNL will perform neutronics and thermal hydraulics analyses to determine key core design parameters, such as fuel pin and coolant channel diameters, pitch-to-diameter ratio, fuel packing fraction, etc. In addition, the reactivity coefficients (temperature, coolant voiding, etc.) will be characterized, and the preferred core volume, shape, power density, etc. will be determined. The INL will develop a Monte Carlo core model to study key reactor physics parameters including fuel loading, temperature and void reactivity coefficients, and other parameters of interest. ANL will contribute to the neutronic analyses efforts using their diffusion theory codes. In collaboration between INL, ANL, and ORNL, a commonly agreed baseline liquid salt-cooled concept will be defined during FY-05.

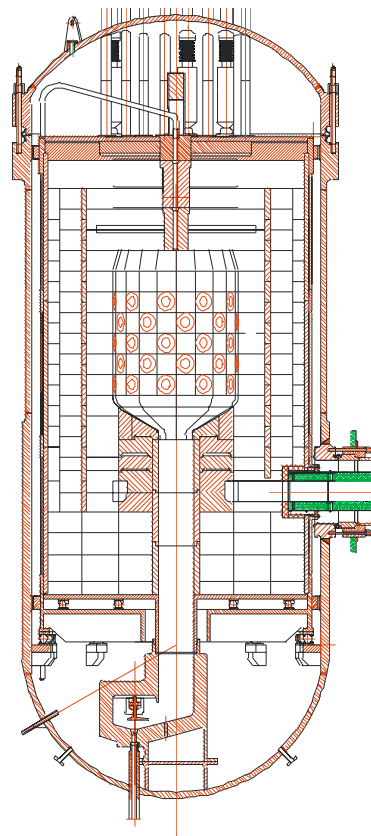


Figure A1.15. Schematic diagram of HTR-10 core and vessel.

**A1.3.4.4.3 Liquid Salt Selection.** ORNL will evaluate the neutronic implications of candidate salts, such as absorption and moderation values, activation, transmutation, etc. In addition, ORNL will survey previous data and experience on use of liquid salts, including those used in the ARE, Medium Power Reactor Experiment, and Molten Salt Breeder Reactor programs, to assess the current knowledge base of thermo-physical and chemical properties and performance and assess the status of molten salt phase diagram modeling for candidate salts.

## A1.3.5 Public Outreach, Analysis, and Decision Support

The public outreach scope is to plan, develop, and perform the work required to begin early and continuing communication with local and regional stakeholders regarding the NGNP. Current analysis of the range of stakeholders and concerns suggest five primary project objectives as summarized in Figure A1.16.

	Local	Regional	National	International
Don't harm the environment	Air, water, land, fish, parks Transportation Why here?		Sustainable energy + environment Nuclear in energy policy context Nuclear in climate change context	
Address regional values	Conserve and protect water + water rights Provide local/regional involvement, oversight, benefits Obtain self-sufficiency (energy, water)			
Aspire to world-class standards	Gen IV objectives and oversight IAEA standards and methodologies Impact on/from other international issues			
Respect people and ideas	Inform, listen, learn, consult, involve Invite diverse viewpoints and publics Show how feedback and concerns are addressed			
Enable employees to be best ambassadors	Increase understanding Obtain feedback Extended networks of friends, family, colleagues			
Increase coordination with other decisions (ICP, RPS, AFCI, YMP, FSV) to earlier understand concerns and values, build relationships, maximize consistency, and avoid counterproductive precedents				

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Figure A1.16. Synopsis of objectives, issues, and range of stakeholders.

The approach will be to:

1. Establish a multi-disciplinary team including members from the INL, academia, and industry.
2. Develop an initial stakeholder list by conducting a two-stage analysis. Identify risks and mitigation strategies.
3. Plan and hold meetings with stakeholders to attain an understanding of the public's perceptions, values, beliefs, concerns, and support for the NGNP. Use a combination of "public meetings" sponsored by public affairs, our focus groups, and individual interviews. Collect and analyze feedback.
4. The stakeholder analysis will be a "living document," periodically updated and discussed with DOE-NE to apprise them of progress and maintain consistency of messages and expectations, etc.

5. Establish an approach and network for future dialogue with the public to ensure strong continuing support for the NGNP, by designing and implementing missions consistent with stakeholder concerns as much as possible.
6. Stay engaged with DOE-NE. Document results and propose support for continuing structured dialogue.

## A1.4 10-YR PROJECT COST AND SCHEDULE

### A1.4.1 10-yr Project Budget

Table A1.5 shows the NGNP budget required to support NGNP initial operation in 2017. The budget figures represented below are initial pre-design estimates, which may change when pre-conceptual designs are completed such that Research and Development can be focused on a specific core design, a project integrator is selected, and engineering estimates can be made. No contingencies have been added to the estimates below.

Table A1.5. NGNP annual budget profile.

NGNP Budget Profile (2017)																
Activity	FY-03	FY-04	FY-05*	FY-06	FY-07	FY-08	FY-09	FY-10	FY-11	FY-12	FY-13	FY-14	FY-15	FY-16	FY-17	Total
<b>Research and Development</b>																
Fuel Development and Qualification	6,000															
Materials Testing and Qualification	480															
Design Methods and Validation	1,200															
Design and Trade Studies																
Public Outreach																
<b>NGNP Subtotal</b>	<b>7,680</b>															
Hydrogen Development	3,000															
<b>DOE SHARE SUBTOTAL</b>	<b>10,680</b>															
<b>INDUSTRY SHARE</b>																
<b>Total</b>	<b>10,680</b>															

\* Includes AFCL, carryover funding and GA funding

Figure A1.17 provides a graphic profile of R&D funding needs against the assumed project time line. Note the major portion of R&D work is completed prior to completion of Final Design.

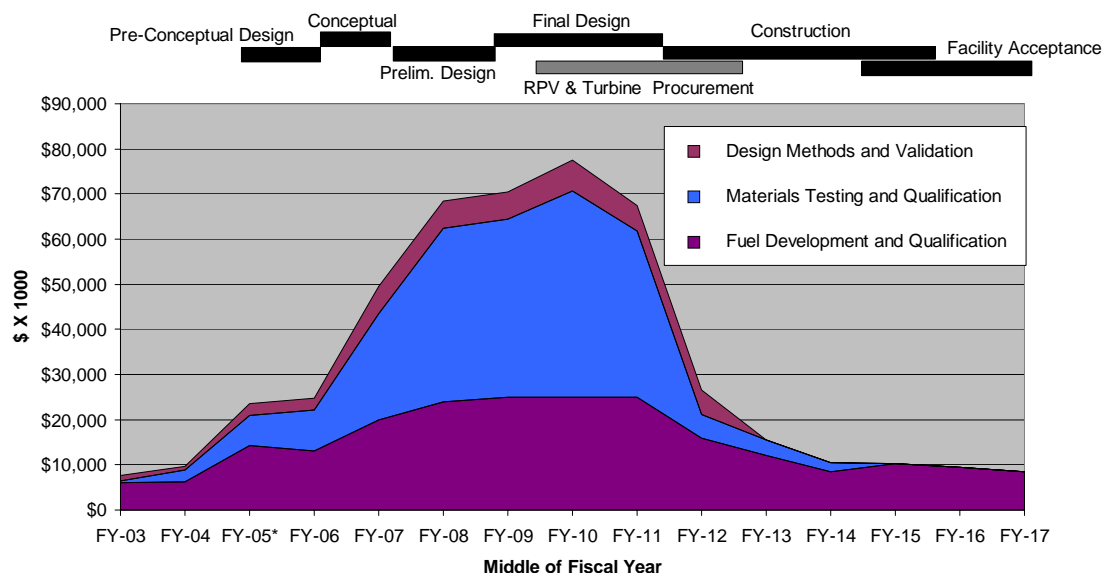


Figure A1.17. NGNP R&D funding profile.

### **A1.4.2 10-yr Project Schedule**

The NGNP Summary Level Schedule is shown in Figure A1.18. The schedule shows the major activities under each of the major research and development areas at the top of the schedule and the assumed design and construction activities are shown in the lower half. The schedule has been prepared from the latest R&D program plans or Field Work Proposals for Fuel Development, Materials Selection and Qualification, and Design Methods and Safety. Energy Conversion is still in planning and a placeholder has been inserted. The following assumptions were made in preparation of the schedule:

- The Design and Construction schedule will follow the principals of DOE O 413.3-1 and an early Critical Decision 1A (CD-1) will be sought in order to finalize the DOE Acquisition Strategy.
- Items shown under the Design and Construction (including the Environmental Impact Statement and Safety Analysis) will be performed by or controlled under the Integrator through industry contracts as determined in the Acquisition Strategy.
- An industry partner will be the NRC licensee.
- The Reactor Pressure Vessel will be needed approximately 12 to 18 month into construction and a 38 to 40 month procurement schedule is needed for the RPV.



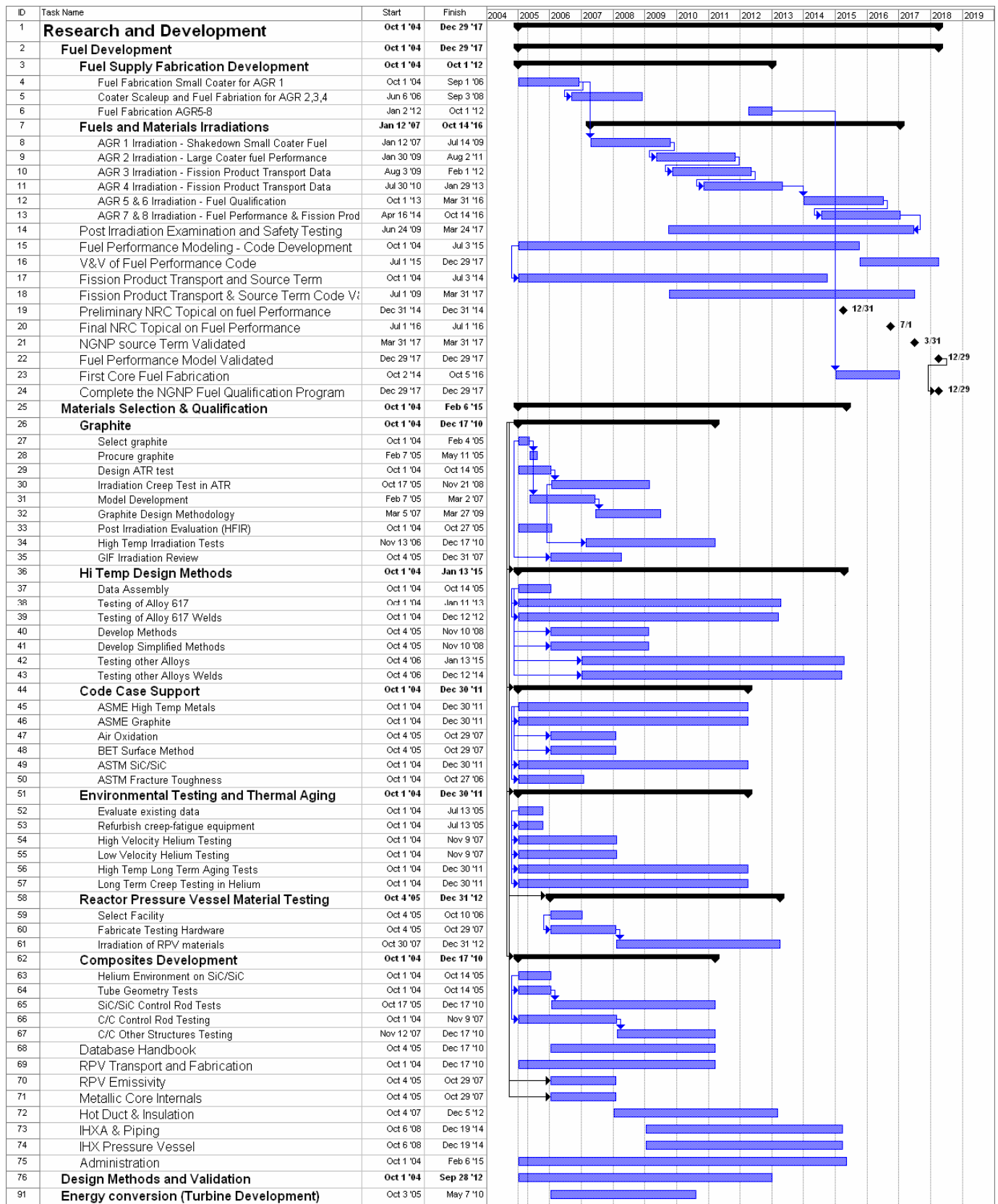


Figure A1.18. NGNP summary schedule.



Note that when the NGNP Acquisition Strategy is developed and approved it is expected that there will be changes required in order for the current R&D planning to correlate with the Integrator's R&D plan. At that time, some issues will need to be resolved. Some of the major issues with the current planning are as follows:

- Fuel qualification irradiations ATR-5 and ATR-6 will be completed in 2015-16. Therefore, the first core load of fuel must be fabricated in parallel with the final qualification irradiations and PIE.
- A commercial fuel manufacturing facility needs to be considered in the overall planning.
- The assumptions for the RPV procurement require that Materials R&D and vessel design will need to be completed by the end of FY-09 in order to support the 2017 schedule above. The current R&D schedule will need to be compressed and funding will have to be accelerated.
- Design Methods and Safety R&D schedule and funding will also need to be accelerated to support Preliminary Design, the preliminary safety analysis and the NRC construction permitting process.

### **A1.4.3 10-yr Project Milestones**

The Major milestones, as outlined in the Office of Nuclear Energy, Science and Technology Program Plan GPRA Unit 14: "Develop New Nuclear Generation Technologies," are as follows:

FY 2006: Complete the Pre-conceptual Design (CD-0)

FY 2007: Complete the Conceptual Design (CD-1)

FY 2009: Complete the Preliminary Design (CD-2)

FY 2012: Start Construction of the NGNP (CD-3)

FY 2017: Complete the NGNP Fuel Qualification Program

FY 2017: Obtain the NRC Operating License

FY 2017: Begin NGNP Operations (CD-4)

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